May 2, 1995

Mr. J. V. Parrish (Mail Drop 1023)
Vice President Nuclear Operations
3000 George Washington Way
Washington Public Power Supply System
P.O. Box 968
Richland, Washington 99352-0968

SUBJECT: ISSUANCE OF AMENDMENT FOR THE WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2 (TAC NOS. M87076 AND M88625)

Dear Mr. Parrish:

The Commission has issued the enclosed Amendment No. 137 to Facility Operating License No. NPF-21 for WPPSS Nuclear Project No. 2. The amendment consists of changes to the Facility Operating License and Technical Specifications (TSs) in response to your application dated July 9, 1993, as supplemented by letters of October 8 and October 25, 1993, January 6, January 6, February 2, May 3, May 13, September 26, and October 12, 1994.

The amendment increases the authorized maximum power level of the reactor from the current limit of 3323 megawatts thermal (MWt) to 3486 MWt. The amendment also modifies TSs to incorporate the increased power limit in the plant operating limits. This request is in accordance with the generic power uprate program for boiling-water reactors (BWRs) established by the General Electric Company and approved by the NRC staff in a letter dated September 30, 1991.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly <u>Federal Register</u> notice.

Sincerely,

Original Signed By

James W. Clifford, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures:
1. Amendment No. 137 to NPF-21
2. Safety Evaluation

cc w/encls: See next page

*For previous concurrences see attached ORCs

<u>DISTRIBUTION</u>:
Docket File
CGrimes, O11E22
Region IV
KPerkins, WCFO
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RJones
RWessman
AThadani
LHurley, RIV
JClifford
ACRS (4), TWFN
PUBLIC
EPeyton
GHill (2), T5C3
OC/LFDCB, T9E10
PDIV-2/RF
RIV, WCFO (4)
CMcCracken
CMiller
RZimmerman
RBarrett

DOCUMENT NAME: WNP87076.AMD

OFC	LA/DRPW	PDIV-2:PM	TECH ED	BC:SICB*	BC:SPLB*	C:SRXB*	C:TERB*	BC:EMEB*
NAME	EPeyton	JClifford	RSanders	JWermiel	CMcCracken	RJones	CMiller	RWessman
DATE	4/26/95	4/26/95	3/14/95	3/27/95	3/21/95	3/21/95	3/24/95	3/24/95

OFC	C:SCCB*	PD:PDIV-2	OGC*	D:DRPW	#2 ADT	#3 ADP 4/26	NRR:DIR
NAME	RBarrett	WBateman	SUttal	EAdensam	AThadani	RZimmerman	WRussell
DATE	3/22/95	4/28/95	4/7/95	4/26/95	4/26/95	4/27/95	4/ /95

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PDR ADOCK 05000397
P PDR

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 2, 1995

Mr. J. V. Parrish (Mail Drop 1023)
Vice President Nuclear Operations
3000 George Washington Way
Washington Public Power Supply System
P.O. Box 968
Richland, Washington 99352-0968

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Sincerely,

James W. Clifford

James W. Clifford, Senior Project Manager
Project Directorate IV-2
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-397

Enclosures: 1. Amendment No. 137 to NPF-21
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

DOCKET NO. 50-397

NUCLEAR PROJECT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 137
License No. NPF-21

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Washington Public Power Supply System (licensee) dated July 9, 1993, as supplemented by letters of October 8, and October 25, 1993, January 6, January 6, February 2, May 3, May 13, September 26, and October 12, 1994, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraphs 2.C.(1) and 2.C.(2) of Facility Operating License No. NPF-21 are hereby amended to read as follows:*

---

* Page 3 is attached, for convenience, for the composite license to reflect this change. Please remove page 3 of the existing license and replace with the attached page.

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3486 megawatts thermal). Items in Attachment 1 shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 137 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This amendment is effective as of the date of issuance and is to be implemented prior to startup from the 1995 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

William T. Russell

William T. Russell, Director
Office of Nuclear Reactor Regulation

Attachments: 1. Page 3 of License
2. Changes to the Technical Specifications

Date of Issuance: May 2, 1995

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source of special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at reactor core power levels not in excess of full power (3486 megawatts thermal). Items in Attachment 1 shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 137 TO FACILITY OPERATING LICENSE NO. NPF-21

DOCKET NO. 50-397

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE	INSERT
ii	ii
xx(a)	xx(a)
1-5	1-5
1-6	1-6
2-4	2-4
3/4 2-2	3/4 2-2
3/4 2-5	3/4 2-5
3/4 2-6	3/4 2-6
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
3/4 3-4	3/4 3-4
3/4 3-5	3/4 3-5
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16
3/4 3-24	3/4 3-24
3/4 3-43	3/4 3-43
3/4 3-55	3/4 3-55
3/4 4-3a	3/4 4-3a
3/4 4-7	3/4 4-7
3/4 4-18	3/4 4-18
3/4 4-19	3/4 4-19
3/4 4-20	3/4 4-20
3/4 4-21	3/4 4-21
3/4 4-21a	3/4 4-22
3/4 4-23	3/4 4-23
3/4 5-3	3/4 5-3
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-4	3/4 6-4
3/4 6-5	3/4 6-5
3/4 6-6	3/4 6-6
3/4 6-33	3/4 6-33
3/4 6-44	3/4 6-44
3/4 7-8	3/4 7-8

REMOVE	INSERT
B 3/4 2-5	B 3/4 2-5
B 3/4 2-6	B 3/4 2-6
B 3/4 3-1	B 3/4 3-1
B 3/4 3-3	B 3/4 3-3
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6	---
B 3/4 4-7	---
B 3/4 5-1	B 3/4 5-1
B 3/4 5-2	B 3/4 5-2
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
B 3/4 6-3	B 3/4 6-3
B 3/4 6-4	B 3/4 6-4
6-21	6-21

INDEX

DEFINITIONS

SECTION

1.0 DEFINITIONS

DEFINITIONS

SECTION

DEFINITIONS (Continued)

INDEX

LIST OF FIGURES

OPERABLE - OPERABILITY

1.28 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

1.29 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

1.30 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation as (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.31 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall, or vessel wall.

1.31a Pa (psig) is $\geq$ the calculated peak containment internal pressure related to design basis accidents, and is equal to 38 psig.

PRIMARY CONTAINMENT INTEGRITY

1.32 PRIMARY CONTAINMENT INTEGRITY shall exist when:

a. All primary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows, or O-rings, is OPERABLE.

PROCESS CONTROL PROGRAM

1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61 and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.34 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3486 MWt.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until deenergization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping, or total steps such that the entire response time is measured.

REPORTABLE EVENT

1.37 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

SECONDARY CONTAINMENT INTEGRITY

1.39 SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All secondary containment penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position.

b. All secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux - High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power - High		
1) Flow Biased	≤ 0.58W + 59%, with a maximum of	≤ 0.58W + 62%, with a maximum of
2) High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 114.9% of RATED THERMAL POWER
c. Fixed Neutron Flux - High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	N.A.	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 1060 psig	≤ 1074 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 13.0 inches above instrument zero	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 10.0% closed	≤ 12.5% closed
6. DELETED		

*See Bases Figure B 3/4 3-1.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) for each type of fuel shall not exceed the limits specified in the Core Operating Limits Report.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the limits specified in the Core Operating Limits Report, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits specified in the Core Operating Limits Report.

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-upscale scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

TRIP SETPOINT	ALLOWABLE VALUE
$S \leq (0.58W + 59\%)T$	$S \leq (0.58W + 62\%)T$
$S_{RB} \leq (0.58W + 50\%)T$	$S_{RB} \leq (0.58W + 53\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 108.5 million lbs/h.
T = Lowest value of the ratio of FRACTION OF RATED THERMAL POWER divided by the MAXIMUM FRACTION OF LIMITING POWER DENSITY. T is always less than or equal to 1.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the APRM flow biased simulated thermal power-upscale scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint value(*) within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and the MFLPD for each class of fuel shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-upscale scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with MFLPD greater than or equal to FRTP.

*With MFLPD greater than the FRTP during power ascension up to 90% of RATED THERMAL POWER, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that APRM readings are greater than or equal to 100% times MFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

3/4.2.6 POWER/FLOW INSTABILITY

LIMITING CONDITION FOR OPERATION

3.2.6 Operation with THERMAL POWER/core flow conditions which lay in Region A of Figure 3.2.6-1 is prohibited.

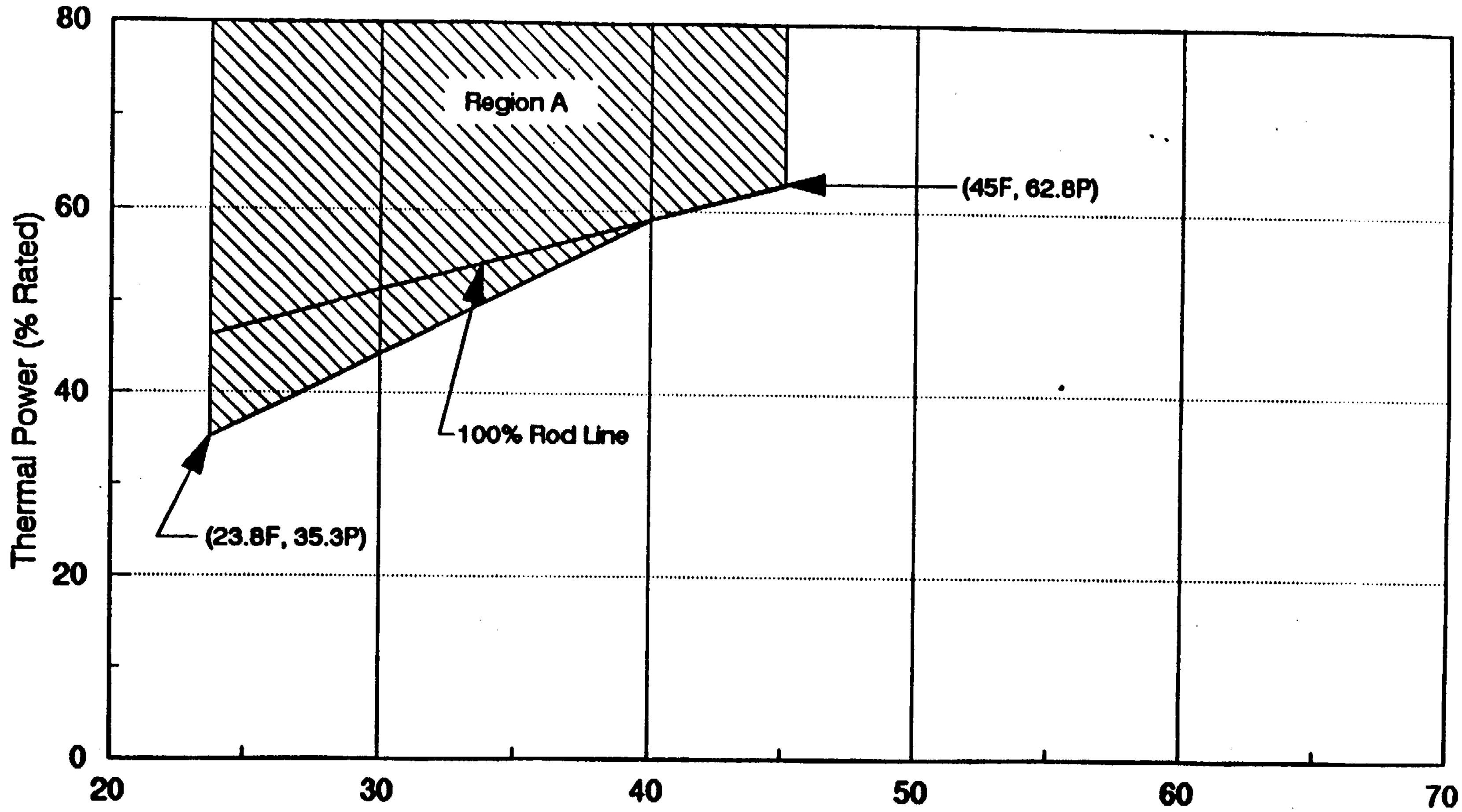
APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than 35.3% of RATED THERMAL POWER and core flow is less than or equal to 45% of rated core flow.

ACTION:

With THERMAL POWER/core flow conditions which lay in Region A of Figure 3.2.6-1, then as soon as practical, but in all cases within 15 minutes, initiate a MANUAL SCRAM.

SURVEILLANCE REQUIREMENTS

4.2.6 The THERMAL POWER/core flow conditions shall be verified to lay outside Region A of Figure 3.2.6-1 once per 24 hours when operating in the region of APPLICABILITY.



CORE FLOW (% RATED)
Operating Region Limits of Specification 3.2.6
Figure 3.2.6-1

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.7 STABILITY MONITORING - TWO LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.2.7 The stability monitoring system shall be operable* and the decay ratio of the neutron signals shall be less than .75 when operating in the region of APPLICABILITY.

APPLICABILITY: OPERATIONAL CONDITION 1, with two recirculation loops in operation and THERMAL POWER/core flow conditions which lay in Region C of Figure 3.2.7-1.

ACTION:

a. With decay ratios of any two (2) neutron signals greater than or equal to 0.75 or with two (2) consecutive decay ratios on any single neutron signal greater than or equal to 0.75:

   As soon as practical, but in all cases within 15 minutes, initiate action to reduce the decay ratio by either decreasing THERMAL POWER with control rod insertion or increasing core flow with recirculation flow control valve manipulation. The starting or shifting of a recirculation pump for the purpose of decreasing decay ratio is specifically prohibited.

b. With the stability monitoring system inoperable and when operating in the region of APPLICABILITY:

   As soon as practical, but in all cases within 15 minutes, initiate action to exit the region of APPLICABILITY by either decreasing THERMAL POWER with control rod insertion or increasing core flow with recirculation flow control valve manipulation. The starting or shifting of a recirculation pump for the purpose of exiting the region of APPLICABILITY when the stability monitoring system is inoperable is specifically prohibited. Exit the region of APPLICABILITY within one (1) hour.

SURVEILLANCE REQUIREMENTS

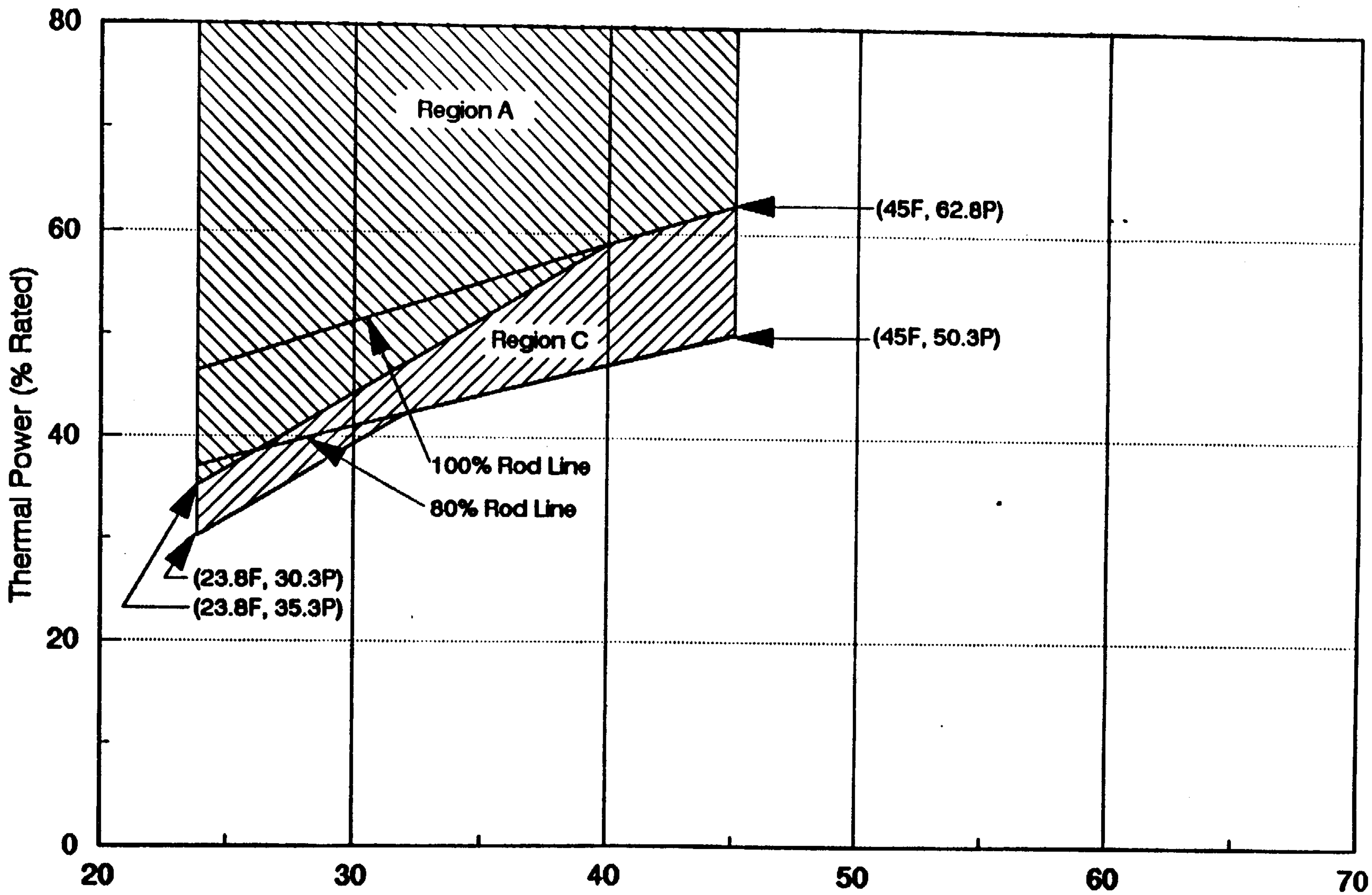
4.2.7.1 The provisions of Specification 4.0.4 are not applicable.

4.2.7.2 The stability monitoring system shall be demonstrated operable* within one (1) hour prior to entry into the region of APPLICABILITY.

4.2.7.3 Decay ratio and peak-to-peak noise values calculated by the stability monitoring system shall be monitored when operating in the region of APPLICABILITY.

---

*Verify that the stability monitoring system data acquisition and calculational modules are functioning, and that displayed values of signal decay ratio and peak-to-peak noise are being updated. Detector levels A and C (or B and D) of one LPRM string in each of the nine core regions (a total of 18 LPRM detectors) shall be monitored. A minimum of four (4) APRMs shall also be monitored.



CORE FLOW (% RATED)
Operating Region Limits of Specification 3.2.7
Figure 3.2.7-1

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.8 STABILITY MONITORING - SINGLE LOOP OPERATION

LIMITING CONDITION FOR OPERATION

3.2.8 The stability monitoring system shall be operable* and the decay ratio of the neutron signals shall be less than .75 when operating in the region of APPLICABILITY.

APPLICABILITY: OPERATIONAL CONDITION 1, with one recirculation loop in operation and THERMAL POWER/core flow conditions which lay in Region C of Figure 3.2.8-1.

ACTION:

a. With decay ratios of any two (2) neutron signals greater than or equal to 0.75 or with two (2) consecutive decay ratios on any single neutron signal greater than or equal to 0.75:

   As soon as practical, but in all cases within 15 minutes, initiate action to reduce the decay ratio by either decreasing THERMAL POWER with control rod insertion or increasing core flow with recirculation flow control valve manipulation. The starting or shifting of a recirculation pump for the purpose of decreasing decay ratio is specifically prohibited.

b. With the stability monitoring system inoperable and when operating in the region of APPLICABILITY:

   As soon as practical, but in all cases within 15 minutes, initiate action to exit the region of APPLICABILITY by decreasing THERMAL POWER with control rod insertion. Exit the region of APPLICABILITY within one (1) hour.

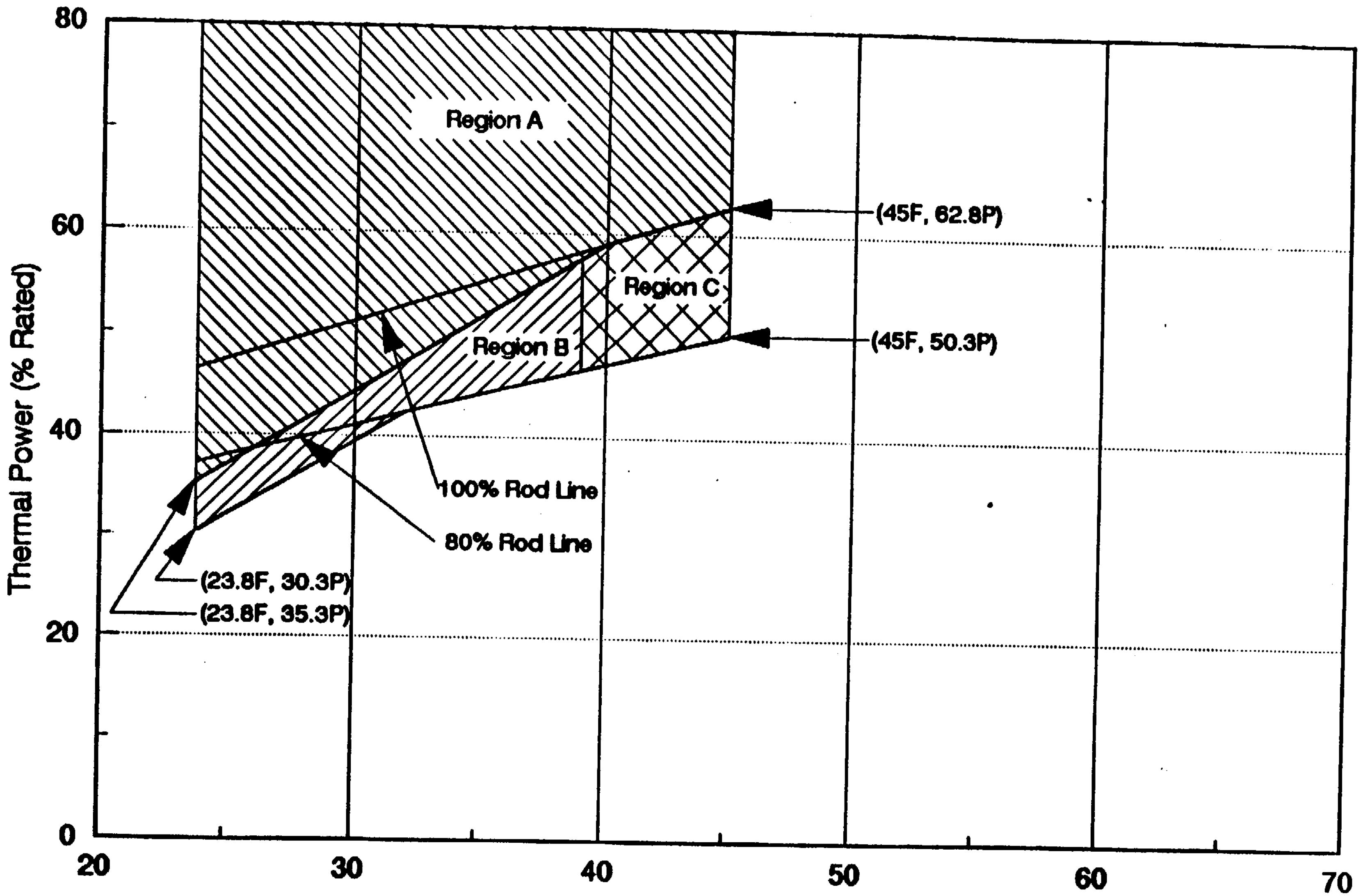
SURVEILLANCE REQUIREMENTS

4.2.8.1 The provisions of Specification 4.0.4 are not applicable.

4.2.8.2 The stability monitoring system shall be demonstrated operable* within one (1) hour prior to entry into the region of APPLICABILITY.

4.2.8.3 Decay ratio and peak-to-peak noise values calculated by the stability monitoring system shall be monitored when operating in the region of APPLICABILITY.

*Verify that the stability monitoring system data acquisition and calculational modules are functioning, and that displayed values of signal decay ratio and peak-to-peak noise are being updated. Detector levels A and C (or B and D) of one LPRM string in each of the nine core regions (a total of 18 LPRM detectors) shall be monitored. A minimum of four (4) APRMs shall also be monitored.



Core Flow (% RATED)
Operating Region Limits of Specification 3.2.8
Figure 3.2.8-1

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
6. DELETED			
7. Primary Containment Pressure - High	1, 2(f)	2(g)	1
8. Scram Discharge Volume Water Level - High			
a. Level Transmitter	1, 2	2	1
	5(h)	2	3
b. Float Switch	1, 2	2	1
	5(h)	2	3
9. Turbine Throttle Valve - Closure	1(i)	4(j)	6
10. Turbine Governor Valve Fast Closure, Valve Trip System Oil Pressure - Low	1(i)	2(j)	6
11. Reactor Mode Switch Shutdown Position	1, 2	1	1
	3, 4	1	7
	5	1	3
12. Manual Scram	1, 2	2	1
	3, 4	2	8
	5	2	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within 1 hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - DELETED

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to a value such that THERMAL POWER is less than 30% of RATED THERMAL POWER, within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within 1 hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS*, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within 1 hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to six hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations are being performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(d) This function shall be automatically bypassed when the reactor mode switch is not in the Run position and reactor pressure < 1060 psig.

(e) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(i) This function shall be automatically bypassed based on turbine first stage pressure when THERMAL POWER is less than 30% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	N.A.
b. Inoperative	N.A.
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	N.A.
b. Flow Biased Simulated Thermal Power - Upscale	6±1**
c. Fixed Neutron Flux - Upscale	≤ 0.09
d. Inoperative	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.55
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.05
5. Main Steam Line Isolation Valve - Closure	≤ 0.06
6. DELETED	
7. Primary Containment Pressure - High	N.A.
8. Scram Discharge Volume Water Level - High	
a. Level Transmitter	N.A.
b. Float Switch	N.A.
9. Turbine Throttle Valve - Closure	≤ 0.06
10. Turbine Governor Valve Fast Closure, Trip Oil Pressure - Low	≤ 0.08#
11. Reactor Mode Switch Shutdown Position	N.A.
12. Manual Scram	N.A.

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION STATEMENTS

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 23 - Be in at least STARTUP within 6 hours.

ACTION 24 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 26 - Lock close or close, as applicable, the affected system isolation valves within 1 hour and declare the affected system inoperable.

TABLE NOTATIONS

*May be bypassed with reactor steam pressure $\leq$ 1060 psig and all turbine stop valves closed.

**When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

#During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) Also actuates the standby gas treatment system.

(c) DELETED

(d) A channel is OPERABLE if 2 of 4 detectors in that channel are OPERABLE.

(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(f) Closes only RWCU system outboard isolation valve RWCU-V-4.

(g) Only valves RHR-V-123A and RHR-V-123B in Valve Group 5 are required for primary isolation.

(h) Manual initiation isolates RCIC-V-8 only and only with a coincident reactor vessel level-low, level 3.

(i) Not required for RHR-V-8 when control is transferred to the alternate remote shutdown panel during operational conditions 1, 2 & 3 and the isolation interlocks are bypassed. When RHR-V-8 control is transferred to the remote shutdown panel under operational modes 1, 2, and 3 the associated key lock switch will be locked with the valve in the closed position. Except RHR-V-8 can be returned to, and operated from, the control room, with the interlocks and automatic isolation capability reestablished in operational conditions 2 and 3 when reactor pressure is less than 135 psig.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level		
1) Low, Level 3	≥ 13.0 inches*	≥ 11.0 inches
2) Low Low, Level 2	≥ -50 inches*	≥ -57 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Main Steam Line		
1) DELETED		
2) Pressure - Low	≥ 831 psig	≥ 811 psig
3) Flow - High	≤ 115.6 psid	≤ 124.6 psid
d. Main Steam Line Tunnel Temperature - High	≤ 164°F	≤ 170°F
e. Main Steam Line Tunnel * Temperature - High	≤ 80°F	≤ 90°F
f. Condenser Vacuum - Low	≥ 23 inches Hg absolute pressure	≥ 24.5 inches Hg absolute pressure
g. Manual Initiation	N.A.	N.A.
2. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Building Vent Exhaust Plenum Radiation - High	≤ 13.0 mR/h	≤ 16.0 mR/h
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Reactor Vessel Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches
d. Manual Initiation	N.A.	N.A.

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	Q	R	1, 2, 3
b. Heat Exchanger Area Temperature - High	N.A.	SA	R	1, 2, 3
c. Heat Exchanger Area Ventilation Δ Temperature - High	N.A	SA	R	1, 2, 3
d. Pump Area Temperature - High				
Pump Room A	N.A	SA	R	1, 2, 3
Pump Room B	N.A.	SA	R	1, 2, 3
e. Pump Area Ventilation Δ Temp. - High				
Pump Room A	N.A.	SA	R	1, 2, 3
Pump Room B	N.A.	SA	R	1, 2, 3
f. SLCS Initiation	N.A.	R	N.A.	1, 2, 3
g. Reactor Vessel Water Level - Low Low, Level 2	N.A.	Q	R	1, 2, 3
h. RWCU/RCIC Line Routing Area Temperature - High	N.A.	SA	R	1, 2, 3
i RWCU Line Routing Area Temperature - High	N.A.	SA	R	1, 2, 3
j. Manual Initiation	N.A.	R	N.A.	1, 2, 3
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	S	Q	R	1, 2, 3
b. RCIC/RHR Steam Line Flow - High	S	Q	R	1, 2, 3
c. RCIC Steam Supply Pressure - Low	N.A.	Q	R	1, 2, 3
d. RCIC Turbine Exhaust Diaphragm Pressure - High	N.A.	Q	R	1, 2, 3
e. RCIC Equipment Room Temperature - High	N.A.	SA	R	1, 2, 3
f. RCIC Equipment Room Δ Temperature - High	N.A.	SA	R	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (Continued)				
g. RWCU/RCIC Steam Line Routing Area Temperature - High	N.A.	SA	R	1, 2, 3
h. Drywell Pressure - High	N.A.	Q	R	1, 2, 3
i. Manual Initiation	N.A.	R	N.A.	1, 2, 3
5. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	S	Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	N.A.	Q	R	1, 2, 3
c. Equipment Area Temperature - High	N.A.	SA	R	1, 2, 3
d. Equipment Area Ventilation Δ Temp. - High	N.A.	SA	R	1, 2, 3
e. Shutdown Cooling Return Flow Rate - High	N.A.	Q	R	1, 2, 3
f. RHR Heat Exchanger Area Temperature - High	N.A.	SA	R	1, 2, 3
g. Manual Initiation	N.A.	R	N.A.	1, 2, 3

TABLE NOTATIONS

* When reactor steam pressure ≥ 1060 psig and/or any turbine stop valve is open.

** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATION and operations with a potential for draining the reactor vessel.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]
1. Turbine Throttle Valve - Closure	2[b]
2. Turbine Governor Valve - Fast Closure	2[b]

---

[a] A trip system may be placed in an inoperable status for up to 2 hours for required surveillance provided that the other trip system is OPERABLE.

[b] This function shall be automatically bypassed when turbine first stage pressure is less than or equal to the pressure equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

TABLE 3.3.4.2-2

END-OF-CYCLE RECIRCULATION PUMP TRIP SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Turbine Throttle Valve-Closure	$\leq$ 5% closed	$\leq$ 7% closed
2. Turbine Governor Valve-Fast Closure	$\geq$ 1250 psig	$\geq$ 1000 psig

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	≤ 0.58W + 48%	≤ 0.58W + 51%
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux Upscale	≤ 0.58W + 50%*	≤ 0.58W + 53%*
b. Inoperative	N.A.	N.A.
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 1 x 10^5 cps	≤ 1.6 x 10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 0.7 cps	≥ 0.5 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ 527 ft 3 in. elevation	≤ 527 ft 5 in. elevation
b. Scram Trip Bypass	N.A.	N.A.
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 108/125 divisions of full scale	≤ 111/125 divisions of full scale
b. Inoperative	N.A.	N.A.
c. Comparator	≤ 10% flow deviation	≤ 11% flow deviation

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	N.A.	S/U(b)(c), Q(c)	Q	1*
b. Inoperative	N.A.	S/U(b)(c), Q(c)	N.A.	1*
c. Downscale	N.A.	S/U(b)(c), Q(c)	Q	1*
2. APRM				
a. Flow Biased Neutron Flux Upscale	N.A.	S/U(b), Q	Q	1
b. Inoperative	N.A.	S/U(b), Q	N.A.	1, 2, 5
c. Downscale	N.A.	S/U(b), Q	Q	1
d. Neutron Flux - Upscale, Startup	N.A.	S/U(b), Q	Q	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	N.A.	S/U(b), W(#)	N.A.	2, 5
b. Upscale	N.A.	S/U(b), W	Q	2, 5
c. Inoperative	N.A.	S/U(b), W	N.A.	2, 5
d. Downscale	N.A.	S/U(b), W	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	N.A.	S/U(b), W(#)	N.A.	2, 5
b. Upscale	N.A.	S/U(b), W	Q	2, 5
c. Inoperative	N.A.	S/U(b), W	N.A.	2, 5
d. Downscale	N.A.	S/U(b), W	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	N.A.	Q	R	1, 2, 5**
b. Scram Trip Bypass	N.A.	Q	N.A.	5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	N.A.	S/U(b), Q	Q	1
b. Inoperative	N.A.	S/U(b), Q	N.A.	1
c. Comparator	N.A.	S/U(b), Q	Q	1

SURVEILLANCE REQUIREMENTS (Continued)

c. Core flow is greater than or equal to 39% of rated core flow when core THERMAL POWER is greater than the limit specified in Figure 3.4.1.1-1.

4.4.1.1.2 With one reactor coolant system recirculation loop not in operation, within no more than 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 25%*** of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 10%*** of rated loop flow:

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant,

b. ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. ≤ 50°F between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements of Specification 4.4.1.1.2b. and c. do not apply when the loop not in operation is isolated from the reactor pressure vessel.

4.4.1.1.3 Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per 18 months by:

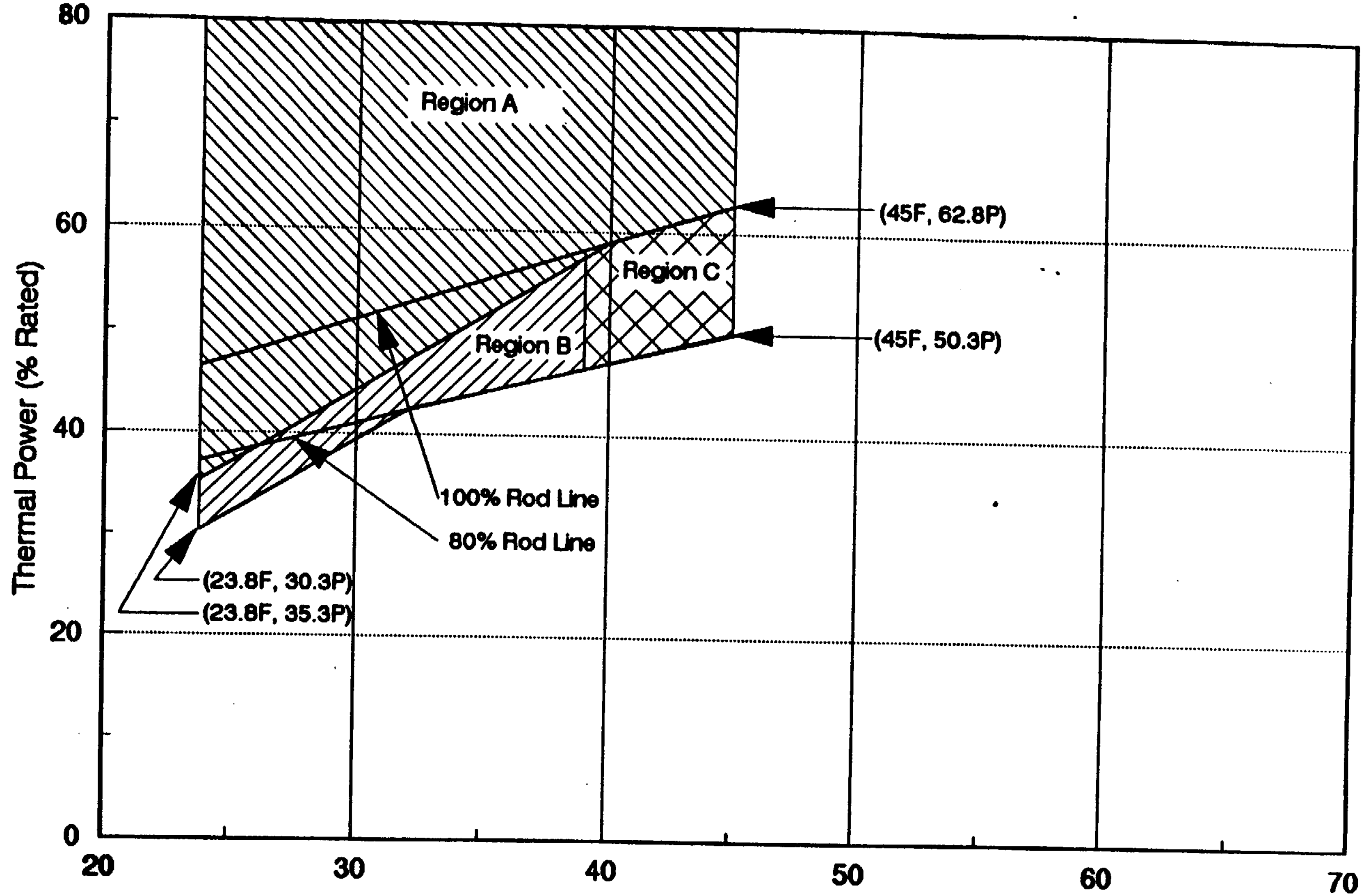
a. Verifying that the control valve fails "as is" on loss of hydraulic pressure (at the hydraulic control unit), and

b. Verifying that the average rate of control valve movement is:

1. Less than or equal to 11% of stroke per second opening, and

2. Less than or equal to 11% of stroke per second closing.

***Final values were determined during Startup Testing based upon actual THERMAL POWER and recirculation loop flow which will sweep the cold water from the vessel bottom head preventing stratification.



Core Flow (%Rated)
OPERATING REGION LIMITS OF SPECIFICATION 3.4.1.1
FIGURE 3.4.1.1-1

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 a) The safety valve function of at least 12 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

2 safety/relief valves @ 1165 psig ±3%
4 safety/relief valves @ 1175 psig ±3%
4 safety/relief valves @ 1185 psig ±3%
4 safety/relief valves @ 1195 psig ±3%
4 safety/relief valves @ 1205 psig ±3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, and 2, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

b) The safety valve function of at least 4 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

2 safety/relief valves @ 1165 psig ±3%
4 safety/relief valves @ 1175 psig ±3%
4 safety/relief valves @ 1185 psig ±3%
4 safety/relief valves @ 1195 psig ±3%
4 safety/relief valves @ 1205 psig ±3%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3, when THERMAL POWER is less than 25% of RATED THERMAL POWER.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 90°F, close the stuck open safety/relief valve(s); if unable to close the open

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Beta and Gamma Activity Determination	At least once per 72 hours	1, 2, 3
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 31 days	1
3. Radiochemical for $\bar{E}$ Determination	At least once per 6 months*	1
4. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1#, 2#, 3#, 4#
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1, 2
5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88	At least once per 31 days	1

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

#Until the specific activity of the primary coolant system is restored to within its limits.

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4.6.1A, 3.4.6.1B, or 3.4.6.1.c* (1) curve A or A′ for hydrostatic or leak testing; (2) curve B or B′ for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period,

c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 80°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown, and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figures 3.4.6.1A, 3.4.6.1.B, or 3.4.6.1.c curves A, A', B, B', or C, as applicable, at least once per 30 minutes.

*Figure 3.4.6.1.c A' and B' curves are effective for less than or equal to 8 EFPY of operation.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1B curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties as required by 10 CFR Part 50, Appendix H in accordance with the NRC approved schedule. The results of these examinations shall be used to update the curves of Figures 3.4.6.1A and Figure 3.4.6.1B.

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 80°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

    1. ≤ 100°F, at least once per 12 hours.

    2. ≤ 90°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

WNP-2 PRESSURE/TEMPERATURE LIMITS

TESTING AND NONNUCLEAR HEATING CURVES "A"&"B"

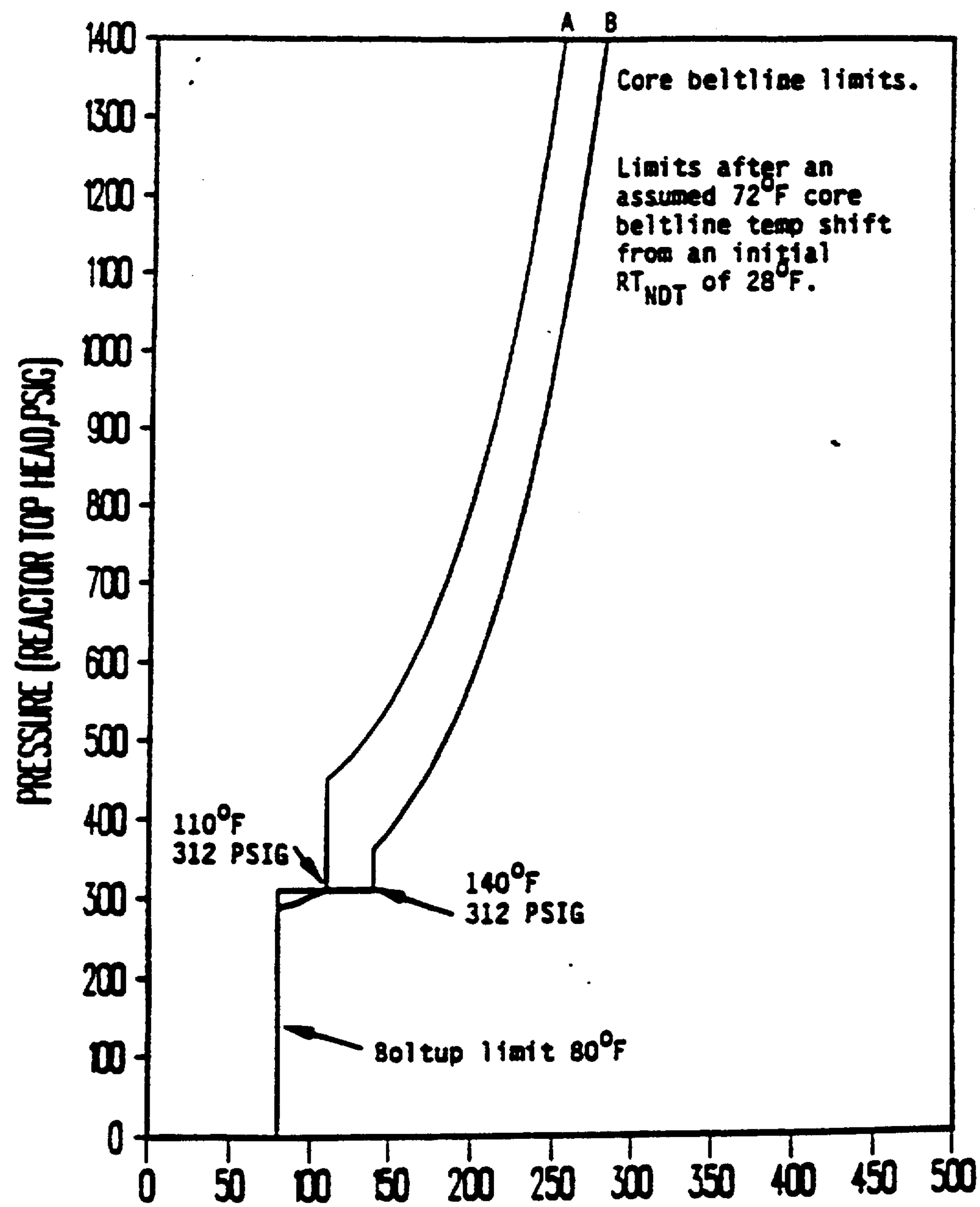


FIGURE 3.4.6.1A

MINIMUM REACTOR METAL TEMPERATURE
TEMPERATURE F

WNP-2 PRESSURE/TEMPERATURE LIMITS FOR 8 EFPY

TESTING AND NONNUCLEAR HEATING CURVES A'& B'

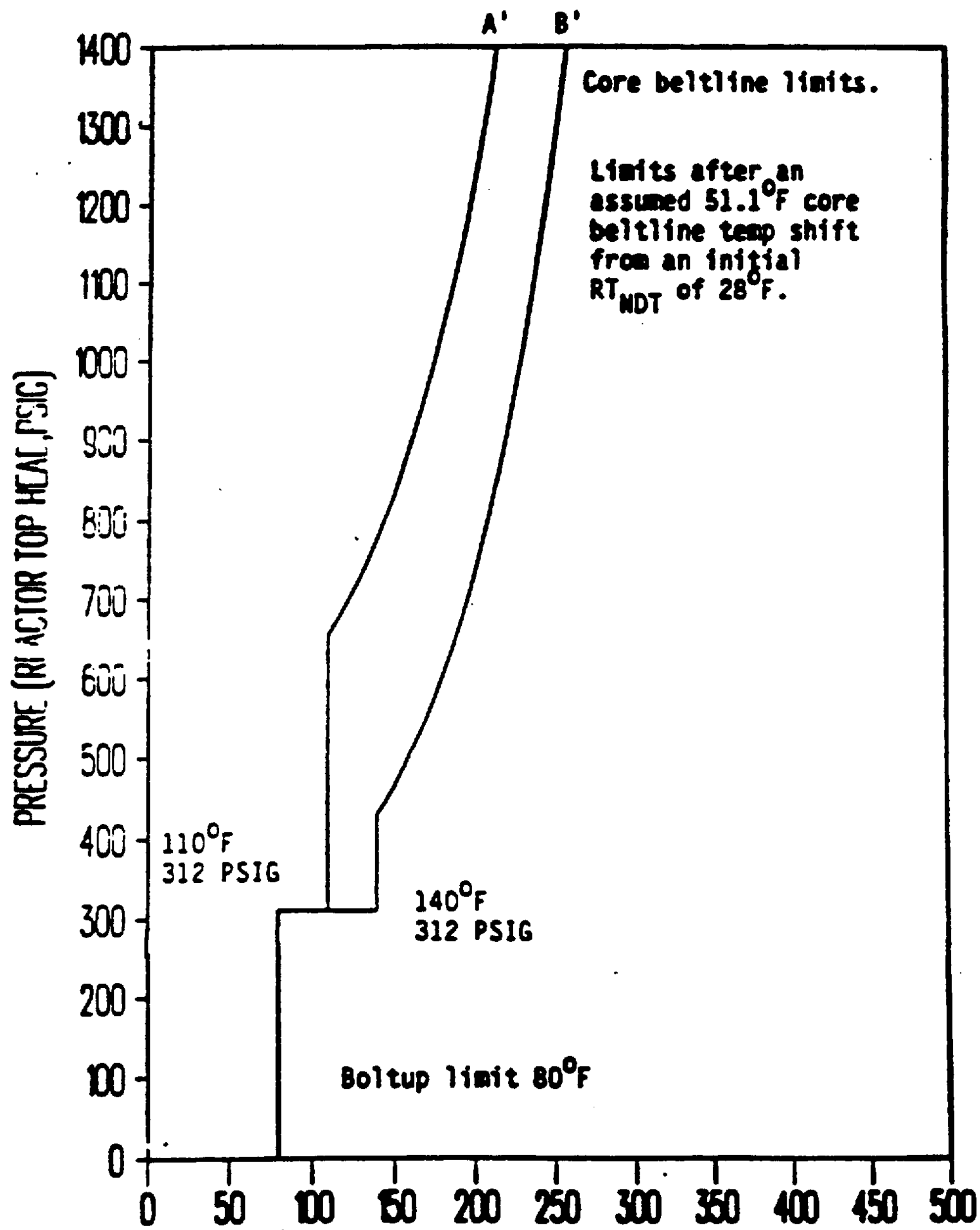


FIGURE 3.4.6.1.c

MINIMUM REACTOR VESSEL METAL TEMPERATURE
TEMPERATURE F

WNP-2 PRESSURE/TEMPERATURE LIMITS

NUCLEAR HEATING CURVE "C"

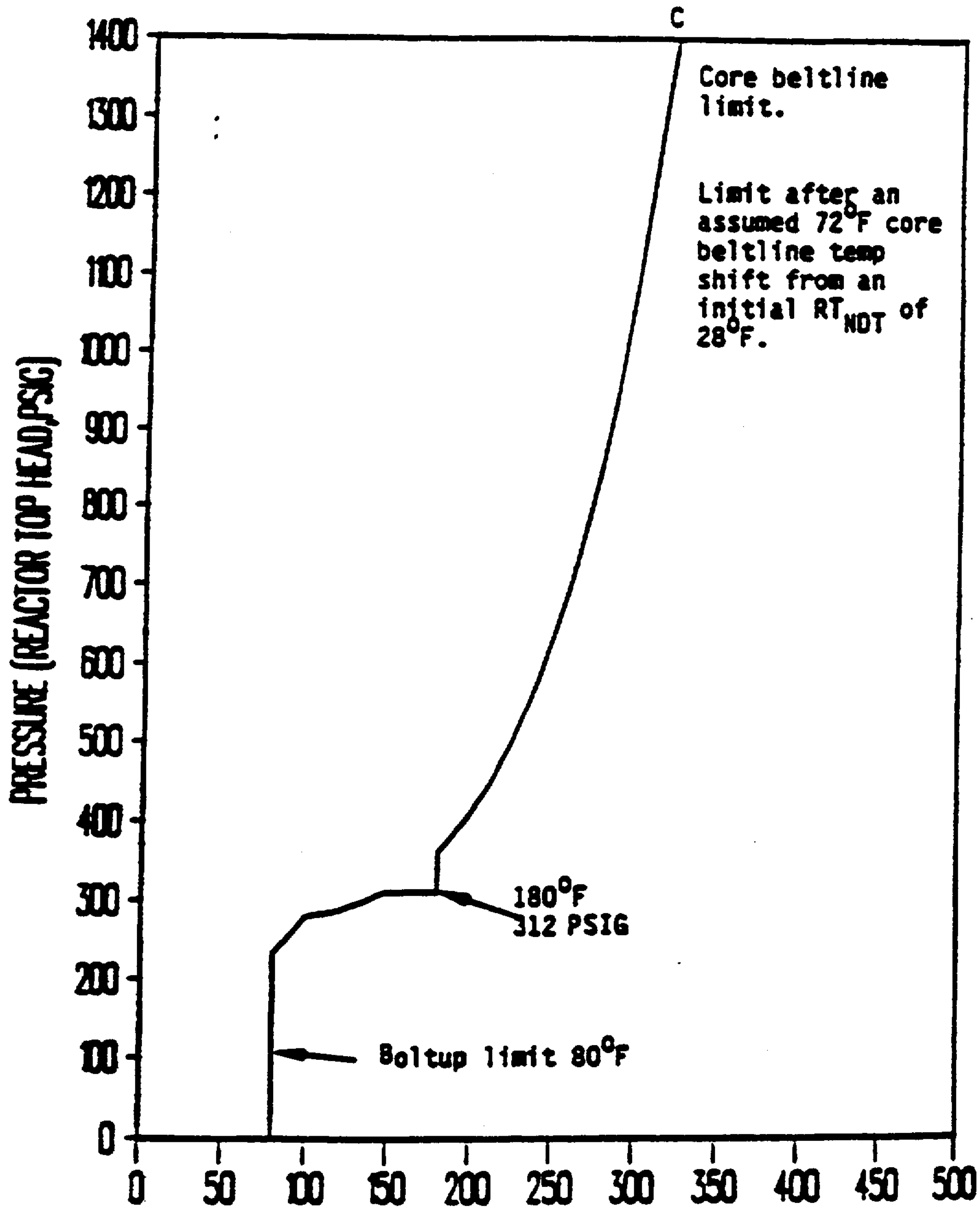


FIGURE 3.4.6.1B

MINIMUM REACTOR METAL TEMPERATURE
TEMPERATURE F

REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1035 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1035 psig, reduce the pressure to less than 1035 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1035 psig at least once per 12 hours.

*Not applicable during anticipated transients.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 and less than or equal to 5 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more MSIVs inoperable:

   1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

      a) Restore the inoperable valve(s) to OPERABLE status, or

      b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 5 seconds when tested pursuant to Specification 4.0.5.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

2) With the LPCS system inoperable and either LPCI subsystems "B" or "C" inoperable, restore at least the inoperable LPCS system or the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

3) Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours*.

e. For ECCS divisions 1 and 2, provided that ECCS division 3 is OPERABLE and divisions 1 and 2 are otherwise OPERABLE:

1. With two of the above required ADS valves inoperable, restore one inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ 128 psig within the next 24 hours.

2. With three or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to ≤ 128 psig within the next 24 hours.

f. In the event an ECCS system is actuated and injects water into the reactor coolant system, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the useage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*Whenever two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

SURVEILLANCE REQUIREMENTS

4.5.1 ECCS divisions 1, 2, and 3 shall be demonstrated OPERABLE by:

a. At least once per 31 days for the LPCS, LPCI, and HPCS systems:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

2. Verifing that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

b. Verifing that, when tested pursuant to Specification 4.0.5, each:

1. LPCS pump develops a flow of at least 6350 gpm at greater than or equal to 128 psid.#

2. LPCI pump develops a flow of at least 7450 gpm at greater than or equal to 26 psid.#

3. HPCS pump develops a flow of at least 6350 gpm at greater than or equal to 200 psid.##

c. For the LPCS, LPCI, and HPCS systems, at least once per 18 months performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

d. For the HPCS system, at least once per 18 months, verifying that the suction is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank low water level signal and on a suppression pool high water level signal.

---

*Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

#Pressure difference between the reactor and the suppression pool air volume.

##Pressure differential above the suction source (Suppression pool or condensate storage tank).

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at P_a, and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

b. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except as provided in Table 3.6.3-1 of Specification 3.6.3.

c. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression chamber is in compliance with the requirements of Specification 3.6.2.1.

* See Special Test Exception 3.10.1.

**Except valves, blind flanges, and deactivated automatic valves which are within the primary containment or other areas administratively controlled to prohibit access for reasons for personnel safety (i.e., radiation and temperature) and are locked, sealed, or otherwise secured in the closed position (1-1/2 inch and smaller valves connected to vents, drains or test connections must be closed but need not be sealed). Valves inside containment shall be verified closed following primary containment de-inerting, but verification is not required more often than once per 92 days. Valves in other administratively controlled areas shall be verified closed during each COLD SHUTDOWN, but verification is not required more often than once per 31 days.

PRIMARY CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to L_a, 0.50 percent by weight of the containment air per 24 hours at P_a.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* (and valves which are hydrostatically leak tested per Table 3.6.3-1), subject to Type B and C tests when pressurized to P_a.

c. *Less than or equal to 11.5 scf per hour for any one main steam line isolation valve when tested at P_t, 25.0 psig.

d. A combined leakage rate of less than or equal to 1 gpm times the total number of ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a.

APPLICABILITY: When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.6.1.1.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, or

b. The measured combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steam line isolation valves* (and valves which are hydrostatically leak tested per Table 3.6.3-1), subject to Type B and C tests exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 11.5 scf per hour for any one main steam line isolation valve, or

d. The measured combined leakage rate for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves,

restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a, and

b. The combined leakage rate for all penetrations and all valves listed in Table 3.6.3-1, except for main steamline isolation valves* and valves which are hydrostatically leak tested per Table 3.6.3-1, subject to Type B and C tests to less than or equal to 0.60 L_a, and

*Exemption to Appendix J of 10 CFR Part 50.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. The leakage rate to less than or equal to 11.5 scf per hour for any one main steam line isolation valve, and

d. The combined leakage rate for all ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972:

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10-month intervals during shutdown at P_a, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

1. Confirms the accuracy of the test by verifying that the supplemental test result, L_c, minus the sum of the Type A and the superimposed leak, L_o, are equal to or less than 0.25 L_a.

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be between 0.75 L_a and 1.25 L_a.

SURVEILLANCE REQUIREMENTS (Continued)

d. Type B and C tests shall be conducted with gas at P_a*, at intervals no greater than 24 months except for tests involving:

   1. Air Locks

   2. Main steam line isolation valves,

   3. Valves pressurized with fluid from a seal system,

   4. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and

   5. Purge supply and exhaust isolation valves with resilient seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

h. ECCS and RCIC containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

i. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.8.1 and 4.6.1.8.2.

j. The provisions of Specification 4.0.2 are not applicable to 24-month or 40 ± 10-month surveillance intervals.

*Unless a hydrosatic test is required per Table 3.6.3-1.

***For those tests conducted during refueling outages, the 24-month interval may be exceeded by no more than 3 months.

PRIMARY CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each primary containment air lock shall be OPERABLE with:

a. The interlock operable and engaged such that both doors cannot be opened simultaneously, and

b. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, and

c. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

a. With the interlock mechanism inoperable:

1. Maintain at least one operable air lock door closed and either return the interlock to service within 24 hours or lock at least one operable air lock door closed.

2. Operation may then continue until the interlock is returned to service provided that one of the air lock doors is verified locked closed prior to each closing of the shield door and at least once per shift while the shield door is open.

3. Personnel passage through the air lock is permitted provided an individual is dedicated to assure that one operable air lock door remains locked at all times so that both air lock doors cannot be opened simultaneously.

4. The provisions of Specification 3.0.4 are not applicable.

b. With one primary containment air lock door inoperable:

1. Maintain at least the OPERABLE air lock door closed and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed immediately prior to each closing of the shield door and at least once per shift while the shield door is open.

*See Special Test Exception 3.10.1.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. The provisions of Specification 3.0.4 are not applicable.

c. With the primary containment air lock inoperable, except as a result of an inoperable air lock door or an inoperable interlock mechanism, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. By verifying interlock operation (i.e., that only one door in each air lock can be opened at a time).

   1. Prior to using the air lock in Operating Conditions 1, 2 and 3 but not required more than once per 6 months,

   2. Following maintenance that could affect the interlock mechanism.

b. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage rate less than or equal to 0.025 L_a when the gap between the door seals is pressurized to 10 psig.

c. By conducting an overall air lock leakage test at P_a, and by verifying that the overall air lock leakage rate is within its limit:

   1. At least once per 6 months###, and

   2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when maintenance had been performed on the air lock that could affect the air lock sealing capability*.

###The provisions of Specification 4.0.3 are not applicable.

*Exception to Appendix J of 10 CFR 50.

TABLE 3.6.3-1 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES

VALVE FUNCTION AND NUMBER	VALVE GROUP(a)	MAXIMUM ISOLATION TIME (Seconds)
d. Other Containment Isolation Valves (Continued)		
Radiation Monitoring		N.A.
PI-V-X72f/1 PI-V-X73e/1		
Transversing Incore Probe System		N.A.
TIP-V-6 TIP-V-7,8,9,10,11(e)		

TABLE NOTATIONS

*But greater than 3 seconds.
#Provisions of Technical Specification 3.0.4 are not applicable.

(a) See Technical Specification 3.3.2 for the isolation signal(s) which operate each group.
(b) Valve leakage not included in sum of Type B and C tests.
(c) May be opened on an intermittent basis under administrative control.
(d) Not closed by SLC actuation signal.
(e) Not subject to Type C Leak Rate Test.
(f) Hydraulic leak test at 1.10 P_a.
(g) Not subject to Type C test. Test per Technical Specification 4.4.3.2.2
(h) Tested as part of Type A test.
(i) May be tested as part of Type A test. If so tested, Type C test results may be excluded from sum of other Type B and C tests.
(j) Reflects closure times for containment isolation only.
(k) During operational conditions 1, 2 & 3 the requirement for automatic isolation does not apply to RHR-V-8. Except that RHR-V-8 may be opened in operational conditions 2 & 3 provided control is returned to the control room, with the interlocks reestablished, and reactor pressure is less than 135 psig.
(l) The isolation logic associated with the reactor recirculation hydraulic control containment isolation valves need not meet single failure criteria for OPERABILITY for a period ending no later than May 15, 1995.

CONTAINMENT SYSTEMS

3/4.6.4 VACUUM RELIEF

SUPPRESSION CHAMBER - DRYWELL VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Seven of the nine pairs of suppression chamber - drywell vacuum breakers shall be OPERABLE and all nine pairs shall be closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more vacuum breakers in up to two pairs of suppression chamber - drywell vacuum breakers inoperable for opening, verify both vacuum breakers of each pair to be closed within two (2) hours.

b. With one or more vacuum breakers in three or more pairs of suppression chamber - drywell vacuum breakers inoperable for opening but known to be closed, restore the inoperable pairs of vacuum breakers such that a minimum of seven pairs are in an OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one suppression chamber - drywell vacuum breaker open, verify the other vacuum breaker in the pair to be closed within 2 hours; restore the open vacuum breaker to the closed position within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one closed position indicator of any suppression chamber - drywell vacuum breaker inoperable:

   1. Verify the other vacuum breaker in the pair to be closed within 2 hours and at least once per 15 days thereafter, or

   2. Verify the vacuum breaker(s) with the inoperable position indicator to be closed by conducting a test which demonstrates that the ΔP is maintained at greater than or equal to 0.5 psi for 1 hour without makeup within 24 hours and at least once per 15 days thereafter.

   3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3/4.6.6 PRIMARY CONTAINMENT ATMOSPHERE CONTROL

DRYWELL AND SUPPRESSION CHAMBER HYDROGEN RECOMBINER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent drywell and suppression chamber hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION: With one drywell and suppression chamber hydrogen recombiner system inoperable, restore the inoperable system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each drywell and suppression chamber hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system warmup test that the minimum recombiner heater outlet temperature increases to greater than or equal to 500°F within 90 minutes.

b. At least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all recombiner operating instrumentation and control circuits.

2. Verifying the integrity of all heater electrical circuits by performing a resistance to ground test within 30 minutes following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

3. Verifying during a recombiner system functional test that, upon introduction of 1% by volume hydrogen in a 140-180 scfm stream containing at least 1% by volume oxygen, that the catalyst bed temperature rises in excess of 120°F within 20 minutes.

4. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure; i.e., loose wiring or structural connections, deposits of foreign materials, etc.

c. By measuring the system leakage rate:

1. As a part of the overall integrated leakage rate test required by Specification 3.6.1.2, or

2. By measuring the leakage rate of the system outside of the containment isolation valves at P_a, on the schedule required by Specification 4.6.1.2, and including the measured leakage as a part of the leakage determined in accordance with Specification 4.6.1.2.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that on each of the below pressurization mode actuation test signals, the train automatically switches to the pressurization mode of operation and the control room is maintained at a positive pressure of 1/8 inch water gauge relative to the outside atmosphere during train operation at a flow rate less than or equal to 1000 cfm:

   a) Drywell pressure-high,

   b) Reactor vessel water level-low, and

   c) Reactor Building exhaust plenum-high radiation.

3. Verifying that the heaters dissipate 5.0 ± 0.5 kW when tested in accordance with ANSI N510-1980.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 while operating the train at a flow rate of 1000 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than 0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the train at a flow rate of 1000 cfm ± 10%.

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression pool and transferring the water to the reactor pressure vessel.**

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

With the RCIC system inoperable, operation may continue provided the HPCS system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

2. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

3. Verifying that the pump flow controller is in the correct position.

b. When tested pursuant to Specification 4.0.5 by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1015 + 20, - 80 psig.*

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* The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

**The ability of automatically taking RCIC suction from the suppression pool is not a requirement for RCIC OPERABILITY until May 17, 1993 or the beginning of the spring 1993 refueling outage when RCIC OPERABILITY is no longer required; whichever occurs first.

slightly increasing the time required for the normal scram to suppress the flux.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

3/4.2.6 POWER/FLOW INSTABILITY

At the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., power shape, bundle power, and bundle flow).

In 1984, GE issued SIL 380 addressing boiling instability and made several recommendations. In this SIL, the power/flow map was divided into several regions of varying concern. It also discussed the objectives and philosophy of "detect and suppress." The SIL recommends that REGION A be bounded by the 100% rod line and REGION C be bounded by the 80% rod line.

The NRC Generic Letter 86-02 discussed both the GE and SIEMENS (then EXXON) stability methodology and stated that due to uncertainties, General Design Criteria 10 and 12 could not be met using available analytical procedures on a BWR. The letter discussed SIL 380 and stated that General Design Criteria 10 and 12 could be met by imposing SIL 380 recommendations in operating regions of potential instabilities. The NRC concluded that regions of potential instability constituted decay ratios of 0.8 and greater by the GE methodology and 0.75 by the SIEMENS methodology which existed at that time.

SIEMENS Power Corporation has recently developed an improved stability computer code STAIF. A topical report (EMF-CC-074P) which describes the STAIF stability code and provides benchmarking against reactor data was submitted to the NRC in 1993. The NRC issued a SER approving the STAIF stability code for establishing stability boundaries on April 14, 1994. In the SER on STAIF the NRC stated the uncertainty in the STAIF code was 20%.

The STAIF stability code has been used to establish the stability region boundaries for WNP-2. The lower boundary of REGION A was defined to assure it bounds a decay ratio of 0.9. REGION C was conservatively defined to bound a decay ratio of 0.75.

The stability REGIONS A and B are shown in Figure 3.2.6-1. REGION A conforms to the recommendations of SIL 380 in that REGION A bounds a calculated decay ratio of 0.9. Operation in REGION A is prohibited. REGION C bounds a decay ratio of 0.75.

3/4.2.7 STABILITY MONITORING - TWO LOOP OPERATION

At the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., rod patterns, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux signal decay ratios should be monitored while operating in this region (Region C).

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio of 0.75 was chosen as the basis for determining the generic region for surveillance to account for the plant to plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a thermal power at which the calculated decay ratio is less than 0.75.

Stability monitoring is performed utilizing the ANNA system. The system shall be used to monitor APRM and LPRM signal decay ratio and peak-to-peak noise values when operating in the region of concern. A minimum number of LPRM and APRM signals are required to be monitored in order to assure that both global (in-phase) and regional (out-of-phase) oscillations are detectable. Decay ratios are calculated from 30 seconds worth of data at a sample rate of 10 samples/second. This sample interval results in some inaccuracy in the decay ratio calculation, but provides rapid update in decay ratio data. A decay ratio of 0.75 is selected as a decay ratio limit for operator response such that sufficient margin to an instability occurrence is maintained. When operating in the region of applicability, decay ratio and peak-to-peak information shall be continuously calculated and displayed. A surveillance requirement to continuously monitor decay ratio and peak-to-peak noise values ensures rapid response such that changes in core conditions do not result in approaching a point of instability.

3/4.2.8 STABILITY MONITORING - SINGLE LOOP OPERATION

The basis for stability monitoring during single loop operation is consistent with that given above for two loop operation. The smaller size of the region of allowable operation, Region C, is due to a limit on the allowed flow above the 80% rodline. When operating above the 80% rodline in single loop operation, the core flow is required to be greater than 39%. Continuous operation in Region B is not permitted. Should Region B be entered the actions required by Technical Specification 3/4.4.1.1 are to be complied with.

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC 30851 P, "Technical Specification Improvement Analyses for BWR Reactor Protection System," as approved by the NRC and documented in the SER (letter to T. A. Pickens from A. Thadani dated July 15, 1987). The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The RPS instrumentation that provides 1) the Turbine Throttle Valve-Closure and 2) Turbine Governor Valve Fast Closure, Valve Trip System Oil Presure - Low trip signals measures first stage turbine pressure to initiate a trip signal. The Load Rejection safety analysis (FSAR 15.2.2) bases initial conditions on rated power and specifies turbine bypass operability at greater than or equal to 30% of rated thermal power. Because first stage pressure can vary depending on operating conditions, the qualifying notes describing when the turbine bypass feature is to be disabled specify a turbine first stage pressure corresponding to less than 30% RTP (turbine first stage pressure is dependent on the operating parameters of the reactor, turbine, and condenser). Therefore, because a value for turbine first stage pressure cannot be precisely fixed and because pressure measurement initiates the trip, the Technical Specification refers to a pressure associated with a specific Rated Thermal Power value rather than a value for pressure.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the safety analyses. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C.-operated valves, a 3-second delay is assumed before the valve starts to move. For A.C.-operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C.-operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 13-second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13-second delay. It follows that checking the valve speeds and the 13-second time for emergency power establishment will establish the response time for the isolation functions. However, to enhance overall system reliability and to monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints, and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971, and NEDO-24222, dated December 1979.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the reactor protection system and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine throttle valves and fast closure of the turbine governor valves.

A fast closure sensor from each of two turbine governor valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine governor valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine throttle valves provides input to one EOC-RPT system; a position switch from each of the other two throttle valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine governor valves and a 2-out-of-2 logic for the turbine throttle valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room. The EOC-RPT System instrumentation that provides a trip signal measures first stage turbine pressure to initiate a trip signal. The safety analysis requiring an EOC-RPT bases initial conditions on rated power and specifies turbine bypass operability at greater than or equal to 30% of rated thermal power. Because first stage pressure can vary depending on operating conditions, the qualifying notes describing when the turbine bypass feature is to disabled specify a turbine first stage pressure corresponding to less than 30% RTP (turbine first stage pressure is dependent on the operating parameters of the reactor, turbine, and condenser). Therefore, because a value for turbine first stage pressure cannot be precisely fixed and because pressure measurement initiates the trip the Technical Specification refers to a pressure associated with a specific Rated Thermal Power value rather than a value for pressure.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190ms, less the time allotted for sensor response, i.e., 10ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., 83ms, and plant preoperational test results.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of Specifications 3/4.1.4, Control Rod Program Controls, 3/4.2, Power Distribution Limits and 3/4.3.1 Reactor Protection System Instrumentation. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The test exception to the weekly Channel Functional Test of the SRM/IRM Detector Not Full In instrumentation noted in Table 4.3.6-1, Control Rod Block Instrumentation Requirements, is intended to avoid cable damage and radiation exposure during operational condition 5 periods when outage work is being done in the under core region. Upon completion of all the work in this area, when access for maintenance or construction efforts is no longer required, the test will be completed per the prescribed frequency within seven days.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with 10 CFR Part 50, Appendix A, General Design Criteria 19, 41, 60, 61, 63, and 64.

The criticality monitor alarm setpoints were calculated using the criteria from 10 CFR 70.24.a.1 that requires detecting a dose rate of 20 Rads per minute of combined neutron and gamma radiation at 2 meters. The alarm setpoint was determined by calculational methods using the gamma to gamma plus neutron ratios from ANSI/ANS 8.3-1979, Criticality Accident Alarm System, Appendix B and assuming a critical mass was formed from a seismic event, with a volume of 6' x 6' x 6' at a distance of 27.7 feet from the two detectors. The calculated dose rate using the methodology is 5.05 R/hr. The allowable value for the alarm setpoint was, therefore, established at 5R/hr.

3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2-hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady-state operation will not exceed small fractions of the dose guidelines of 10 CFR Part 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady-state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron irradiation, E greater than 1 MeV, will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content, and copper content of the material in question, can be predicted using the fluence for 109.2% of original rated power and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure/temperature limit curves, Figures 3.4.6.1A and 3.4.6.1B include predicted adjustments for this shift in RT_{NDT} for the end of life fluence and are effective for 10 EFPY and 8 EFPY, respectively.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figures 3.4.6.1A and 3.4.6.1B shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figures 3.4.6.1A and 3.4.6.1B for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Access to permit inservice inspections of components of the reactor coolant system is in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through Summer 1975.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a.

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

ECCS division 1 consists of the low pressure core spray system and low pressure coolant injection subsystem "A" of the RHR system and the automatic depressurization system (ADS) as actuated by ADS trip system "A". ECCS division 2 consists of low pressure coolant injection subsystems "B" and "C" of the RHR system and the automatic depressurization system as actuated by ADS trip system "B".

The low pressure core spray (LPCS) system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

The LPCS is a primary source of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the LPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

The low pressure coolant injection (LPCI) mode of the RHR system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. Three subsystems, each with one pump, provide adequate core flooding for all break sizes up to and including the double-ended reactor recirculation line break, and for small breaks following depressurization by the ADS.

The surveillance requirements provide adequate assurance that the LPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

ECCS division 3 consists of the high pressure core spray system. The high pressure core spray (HPCS) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system operates over a range of 1210 psid, differential pressure between reactor vessel and HPCS suction source, to 0 psid.

ECCS - OPERATING and SHUTDOWN (Continued)

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to 516/1550/6350 gpm at differential pressures of 1160/1130/200 psig. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety/relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls seven selected safety/relief valves although the safety analysis only takes credit for five valves. It is therefore appropriate to permit two valves to be out-of-service for up to 14 days without materially reducing system reliability.

3/4.5.3 SUPPRESSION CHAMBER

The suppression chamber is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS, and LPCI systems in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL CONDITION 1, 2, or 3 is required by Specification 3.6.2.1.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the calculated peak accident pressure. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation valve leak testing and testing the airlocks after each opening.

3/4.6.1.3 PRIMARY CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR Part 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIVs such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

MSIV LEAKAGE CONTROL SYSTEM (Continued)

Design specifications require the system to accommodate a leak rate of five times the Technical Specification leakage allowed for the MSIVs while maintaining a negative pressure downstream of the MSIVs. The allowed leakage value per each valve is 11.5 scfm, or a total of 230 scfh (3.8 scfm).[a] When corrected for worst case pressure, temperature and humidity expected to be seen during surveillance testing conditions, the flow would never exceed an indicated value (uncorrected reading from local flow instrumentation) of 5 cfm. The 30 cfm acceptance criterion provides significant margin to this design basis requirement and provides a benchmark for evaluating long term blower performance. The Technical Specification limit for pressure of -17" H_2O W.C. was also established based on a benchmark of the installed system performance capability. This -17" H_2O W.C. provides assurance that the negative pressure criterion can be met.

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum calculated pressure in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the calculated containment peak pressure does not exceed the design pressure of 45 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 2 psid. The limit of 1.75 psig for initial positive containment pressure will limit the peak pressure to be less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The 24-inch and 30-inch drywell and suppression chamber purge supply and exhaust isolation valves are required to be closed during plant operation except as required for inerting, de-inerting and pressure control. Until all the drywell and suppression chamber valves have been qualified as capable of closing within the times assumed in the safety analysis, they shall not be open more than 90 hours in any consecutive 365 days. Valves not capable of closing from a full open position during a LOCA or steam line break accident shall be blocked so as not to open more than 70°.

(a) Letter, G02-75-238, dated August 18, 1975, NO Strand (SS) to OD Parr (NRC), "Response to Request for Information Main Steam Isolation Valve Leakage Control System"

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM (Continued)

The time limit on use of the drywell and suppression chamber purge lines is not restricted when using the 2-inch purge supply and exhaust isolation valves since the 2-inch valves will close during a LOCA or steam line break accident and therefore the SITE BOUNDARY dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during PURGING operations. The design of the 2-inch purge supply and exhaust isolation valves meets the requirements of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations."

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. Valves with metal to metal seals will be tested on a Type C schedule in accordance with Surveillance 4.6.1.2.d to assure allowable leakage rates are not exceeded. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of those valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2. DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure of 45 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1040 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss-of-coolant accident, the pressure of the liquid must not exceed 45 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant and to be considered is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is below the design pressure of 45 psig. Maximum water volume of 128,827 ft^3 results in a downcomer submergence of 12 ft and the minimum volume of 127,197 ft^3 results in a submergence approximately 4 inches less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to the downcomer submergence, this specification is adequate. The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

DEPRESSURIZATION SYSTEMS (Continued)

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 90°F results in a water temperature of approximately 145°F immediately following blowdown which is below the 200°F used for complete condensation via quencher devices. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus, there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependency on containment overpressure for post-LOCA operations.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak bulk temperature of the suppression pool is maintained below 200°F during any period of relief valve operation with sonic conditions at the discharge exit for quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety/relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety/relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety/relief valve to assure mixing and uniformity of energy insertion to the pool.

3/4.6.3 PRIMARY CONTAINMENT ISOLATION VALVES

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures for those isolation valves designed to close automatically that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

CORE OPERATING LIMITS REPORT (Continued)

6.9.3.2 The analytical methods used to determine the core operating limits shall be those topical reports and those revisions and/or supplements of the topical report previously reviewed and approved by the NRC, which describe the methodology applicable to the current cycle. For WNP-2 the topical reports are:

1. ANF-1125(P)(A), and Supplements 1 and 2, "ANFB Critical Power Correlation," April 1990
2. Letter, R. C. Jones (NRC) to R. A. Copeland (ANF), "NRC Approval of ANFB Additive Constants for ANF 9x9-9X BWR Fuel," dated November 14, 1990
3. ANF-NF-524(P)(A), Revision 2 and Supplements 1 and 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," November 1990
4. ANF-913(P)(A), Volume 1, Revision 1 and Volume 1, Supplements 2, 3 and 4, "COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analysis," August 1990
5. ANF-CC-33(P)(A), Supplement 2, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50, Appendix K Heatup Option," January 1991
6. XN-NF-80-19(P)(A), Volume 1, Supplements 3 and 4, "Advanced Nuclear Fuel Methodology for Boiling Water Reactors," November 1990
7. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," June 1986
8. XN-NF-80-19(P)(A), Volume 3, Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description," January 1987
9. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," September 1986
10. ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, "Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel," October 1991
11. XN-NF-81-22(P)(A), "Generic Statistical Uncertainty Analysis Methodology," November 1983
12. NEDE-24011-P-A-10-US, "General Electric Standard Application for Reactor Fuel," U.S. Supplement, March 1991
13. NEDE-23785-1-PA, Revision 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology," October 1984
14. NEDO-20566A, "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50 Appendix K," September 1986
15. EMF-CC-074(P)(A), "Volume 1 -- STAIF - A Computer Program for BWR Stability in the Frequency Domain, Volume 2 -- STAIF A Computer Program for BWR Stability in the Frequency Domain, Code Qualification Report," July 1994.

CORE OPERATING LIMITS REPORT (Continued)

6.9.3.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits and accident analysis limits) of the safety analysis are met.

6.9.3.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 137 TO FACILITY OPERATING LICENSE NO. NPF-21

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

NUCLEAR PROJECT NO. 2

DOCKET NO. 50-397

1.0 INTRODUCTION

In its letter dated July 9, 1993 (Reference 1), as supplemented by letters dated October 8, 1993, October 25, 1993, January 6, 1994 (Reference 2), January 6, 1994 (Reference 3), February 2, 1994 (Reference 4), May 3, 1994 (Reference 5), May 13, 1994 (Reference 6), September 26, 1994, and October 12, 1994, the Washington Public Power Supply System (the Supply System, or the licensee) proposed that Facility Operating License No. NPF-21 and Appendix A (Technical Specifications [TSs]) of Facility Operating License No. NPF-21 be amended. The proposed changes would increase the licensed thermal power level of the reactor from the current limit of 3323 megawatts thermal (MWt) to a limit of 3486 MWt. This request conforms to the generic power uprate program for boiling-water reactors (BWRs) established by the General Electric Company (GE) (Reference 7) and approved by the U.S. Nuclear Regulatory Commission (NRC) staff in a letter dated September 30, 1991 (Reference 8).

2.0 DISCUSSION

On December 28, 1990, GE submitted a proposal (Reference 9) to create a generic program to increase the rated thermal power levels of the BWR/4, BWR/5, and BWR/6 product lines by approximately 5 percent. The report contained a proposed outline for individual license amendment submittals and discussed the scope and depth of plant-specific reviews needed and the methodologies to be used in these reviews. In Reference 8, the NRC staff approved the program proposed by GE, on the condition that individual power uprate amendment requests meet certain requirements in the GE document.

The generic BWR power uprate program gives each licensee a consistent means to recover additional generating capacity beyond its current licensed limit, up to the reactor power level used in the original design of the nuclear steam supply system (NSSS). The original licensed power level for most licensees was based on the vendor-guaranteed power level for the reactor. The difference between the guaranteed power level and the design power level is often referred to as stretch power. The design power level is used in determining the specifications for all major NSSS equipment, including the emergency core cooling systems (ECCSs). Therefore, increasing the rated thermal power limits within this program does not violate the design parameters of the NSSS equipment and does not significantly affect the reliability of this equipment.

The Supply System's amendment request to increase the current licensed power level of 3323 MWt to a new limit of 3486 MWt represents an approximate 4.9-percent increase in thermal power with a corresponding 5-percent increase in rated steam flow. The licensee will increase power to the higher level by (1) increasing the core thermal power to increase steam flow, (2) increasing feedwater system flow by a corresponding amount, (3) increasing reactor pressure to ensure adequate turbine control margin, (4) not increasing the current maximum core flow, and (5) operating the reactor along higher flow control lines. This approach is consistent with the BWR generic power uprate guidelines presented in Reference 9. The operating pressure will be increased approximately 15 psi to ensure satisfactory pressure control and pressure drop characteristics for the increased steam flow. The increased core power will be achieved by utilizing a flatter radial power distribution while still maintaining limiting fuel bundles within their constraints.

3.0 EVALUATION

The NRC staff reviewed the request for the WNP-2 power uprate amendment using applicable rules, regulatory guides, sections of the Standard Review Plan, and NRC staff positions. The NRC staff also evaluated the Supply System's submittal for compliance with the generic BWR power uprate program as defined in Reference 9. Detailed discussions of individual review topics follow.

3.1 Reactor Core and Fuel Performance

The NRC staff evaluated the power uprate for its effect on such areas related to reactor thermohydraulic and neutronic performance as power/flow operating map, core stability, reactivity control, fuel design, control rod drives, and scram performance. The NRC staff also considered the effect of power uprate on reactor transients, anticipated transients without scram (ATWS), ECCS performance, and peak cladding temperature for design-basis accident break spectra.

3.1.1 Fuel Design and Operation

The licensee stated in Reference 1 that no new fuel designs would be needed in order to increase power. This is consistent with the information in Reference 7. The plant will continue to comply with fuel operating limits, including the maximum average planar linear heat generation rate (MAPLHGR) and operating limit minimum critical power ratio (OLMCPR) for future reloads. The OLMCPR is determined on a cycle-specific basis from the results of reload analysis, as described in Reference 7. The MAPLHGR and LHGR limits will also be maintained as described in this Reference. The plant-specific safety analysis for WNP-2 is in References 10, 11, and 12. Cycle-specific thermal limits will be included in the core operating limits report (COLR).

3.1.2 Power/Flow Operating Map

The uprated power/flow map includes the operating domain changes for uprated power. The map considers the increased core flow (ICF) range and an uprated extended load line limit (ELLL) analysis. The maximum thermal operating power and maximum core flow correspond to the uprated power and the maximum core flow for ICF. Uprated power has been rescaled so that it is equal to 100-percent rated. The changes to the power/flow operating map are consistent with the previously approved generic descriptions given in Reference 13 and are acceptable to the staff.

3.1.3 Stability

The licensee evaluated the effect of power uprate on core stability issues according to the generic guidelines for power uprate (Reference 7). To determine the effect on core stability, the licensee reviewed the recommendations from GE Service Information Letter (SIL)-380, Revision 1; NRC Bulletin 88-07, Supplement 1; and current efforts of the BWR Owners Group (BWROG), including interim corrective actions (ICAs) recommended by GE and the BWROG. In addition, the licensee revised the power/flow maps on the basis of an updated version of the STAIF code, as discussed in Reference 13. Use of the code was approved by the NRC staff, with limitations, in Reference 14. The power/flow maps proposed by the licensee were determined to preserve a decay ratio of 0.9 or less, which is consistent with NRC Bulletin 88-07, Supplement 1 (Reference 15). Operating within the proposed limits on the power/flow maps is intended to provide adequate margin to ensure thermal-hydraulic stability, and thus maintain the level of protection against power instability that is provided by the STAIF code to presently authorized power conditions.

Ongoing activities by the BWROG and the NRC are addressing ways to minimize the occurrence and potential effects of power oscillations that have been observed for certain BWR operating conditions (as required by General Design Criterion 12 of 10 CFR Part 50 Appendix A). GE has documented information and cautions concerning this possibility in SIL-380 and related communications. The NRC has documented its concerns in NRC Bulletin 88-07 and Supplement 1 to that bulletin. While a more permanent resolution is being developed, the technical specifications and associated implementing procedures proposed by the licensee using the STAIF code will restrict plant operation in the high power, low core flow region of the BWR power/flow operating map. Specific operator actions have been established to provide clear instructions for the possibility that a reactor inadvertently (or under controlled conditions) enters any of the defined regions.

Specific restrictions discussed in Bulletin 88-07 and Supplement 1 apply to uprated operation. Final resolution will continue to proceed as directed by the joint effort of the BWROG and the NRC. This is acceptable to the NRC staff.

3.1.4 Control Rod Drives (CRDs) and CRD Hydraulic System

The CRD system controls gross changes in core reactivity by positioning neutron-absorbing control rods within the reactor. The CRD system is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The licensee evaluated the CRD system at uprated steam flow and dome pressure.

The increase in dome pressure at uprated power will increase the bottom head pressure a corresponding amount. Although the increased pressure will slow rod insertion initially, the reactor pressure will eventually become the primary source of pressure to complete the scram. Hence, the higher reactor pressure will improve scram performance after the initial degradation. Increased reactor pressure has little effect on scram time, and CRD performance during power uprate will conform to current TS requirements, as will be verified through required periodic measurement of individual CRD and group scram times.

Power uprate conditions reduce the operating margin between available and required drive water differential pressure. For CRD insertion and withdrawal, the required minimum differential pressure between the hydraulic control unit (HCU) and the vessel bottom head is 250 psi. The licensee analyzed plant CRD pump and system data, and the CRD pumps were found to have sufficient capacity. The flow required for CRD cooling and driving is assured by the automatic opening of the system flow control valve, thus compensating for the small increase in reactor pressure. The licensee has determined through evaluation that the flow control valves and CRD pumps are capable of operating within their acceptable range with power uprate.

Based on its review of the information provided by the licensee, the NRC staff concludes that the CRD system will continue to perform all its safety-related functions at uprated power with ICF, will function adequately during insert and withdraw modes, and is, therefore, acceptable.

3.2 Reactor Coolant System and Connected Systems

The NRC staff reviewed the mechanical portions of the Supply System's power uprate amendment request to determine the effects of power uprate on the structural and pressure boundary integrity of the piping systems and components, their supports, and reactor vessel and internal components. The NRC staff's review is discussed below.

3.2.1 Nuclear System Pressure Relief

The plant safety/relief valves (SRVs) and reactor scram functions relieve pressure from the nuclear system to prevent overpressurization of the nuclear system during abnormal operating transients. The setpoints for the relief function of the SRVs are increased 15 psi to accommodate the increased dome pressure for power uprate.

The maximum operating steam dome pressure was selected to enable the turbine control valves (TCVs) to operate effectively at the higher steam flow condition corresponding to uprated power. An appropriate increase in the SRV safety setpoints ensures that adequate differences are maintained between operating pressure and safety setpoints (simmer margin), and that the increase in steam dome pressure does not increase the number of unnecessary SRV actuations. The evaluation in Section 3.2.2 of this safety evaluation (SE) discusses the capability of the nuclear boiler pressure relief system to accommodate the power uprate.

3.2.2 Code Overpressure Protection

The design pressure of the reactor pressure vessel (RPV) and reactor coolant pressure boundary (RCPB) remains at 1250 psig. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code allows a peak pressure of 1375 psig (110 percent of the design value), which is the acceptance limit for pressurization events. The limiting pressurization event is a main steam isolation valve (MSIV) closure with a failure of the valve position scram. The licensee's uprate analysis assumes (1) core power is 3702 MWt (106.2 percent of the uprated power of 3486 MWT), (2) end-of-cycle nuclear parameters, (3) six SRVs out of service, (4) no credit for the relief mode of the SRVs, (5) TS scram speed, (6) 3-second MSIV closure time, and (7) initial reactor dome pressure of 1050 psia. The SRV opening pressures were +3 percent above the nominal safety setpoint for the available valves. The analysis resulted in a peak vessel bottom head pressure of 1335 psig. This value remains below the ASME Code limit of 1375 psig. The NRC staff reviewed the results of the licensee's evaluation and finds this acceptable.

3.2.3 Reactor Pressure Vessel and Internals

The licensee evaluated the reactor vessel and internal components considering load combinations that contain reactor internal pressure difference (RIPD), loss-of-coolant-accident (LOCA), safety relief valve (SRV), seismic, annulus pressurization (AP), and fuel lift loads.

The licensee evaluated such LOCA loads as pool swell, condensation oscillation (CO), and chugging for the WNP-2 power uprate and found that the original LOCA analyses remain unchanged because the containment conditions with the power uprate are within the range of test conditions used to define the LOCA dynamic loads. The licensee evaluated the SRV containment dynamic loads that affect the reactor vessel and piping systems. The licensee determined that the increase of SRV loads resulting from the change of SRV setpoints are within the range of the original WNP-2 SRV load definition on the suppression pool boundary. Therefore, the original SRV loads remain bounding for the power uprate condition. The licensee reviewed the original analyses for the AP and found that the mass and energy release rates used for calculation of the original analyzed loads bound the uprated power conditions. On the basis of this review, the NRC staff concurs with the licensee's determination that the LOCA, SRV, and AP design-basis loads remain bounding for the WNP-2 power uprate. There is no change in the maximum allowable core flow for WNP-2 power uprate.

The licensee determined that the fuel lift loads for power uprate conditions are less than those specified for the original design-basis for WNP-2. The licensee determined that this was due primarily to improvements in the analytical model.

The licensee evaluated the stresses and fatigue usage factor for reactor vessel components in accordance with the requirements of the 1971 Edition of the ASME Boiler and Pressure Vessel Code Section III, Subsection NB with Summer 1971 Addenda (Reference 16) to ensure compliance with the WNP-2 original code of record. The load combinations for normal, upset, and faulted conditions were considered in the evaluation. The maximum stresses at the critical locations for the shroud and top guide were summarized in Table 1 of Reference 6. The fatigue usage factors of limiting components calculated for the uprated power level were listed in Table 3-4 of Reference 10. The maximum cumulative usage factor is 0.696 located at the feedwater nozzle based on 40 years of operation. No new assumptions were used in the analysis for the power uprate condition.

The NRC staff reviewed the results of the licensee's evaluation and concludes that the maximum stresses and fatigue usage factor submitted by the licensee are within the code-allowable limits and are, therefore, acceptable.

3.2.4 Control Rod Drive System

The licensee evaluated the adequacy of the WNP-2 control rod drive mechanism (CRDM) in accordance with the ASME Boiler and Pressure Vessel Code Section III, 1968 Edition, up to Winter 1970 Addenda (Reference 17). The licensee found that the limiting component of the CRDM was the indicator tube. The maximum calculated stress was based on a maximum CRD internal water pressure of 1750 psig. This basis is not affected by the power uprate. The licensee calculated a maximum fatigue usage factor of 0.15 for the CRD main flange assuming 40 years of plant operation.

The increase in the reactor dome pressure, operating temperature, and steam flow rate as a result of the power uprate are bounded by the conditions assumed in the GE generic guidelines for the power uprate (Reference 5). The CRDM was originally evaluated for a normal maximum reactor dome pressure of 1060 psig, which is higher than the power uprate dome pressure of 1020 psig. The licensee also stated that the CRDM has been tested at simulated reactor pressure up to 1250 psig, which bounds the analytic limit for the high-pressure scram setpoint of 1086 psig for the power uprate.

Based on its review of the licensee's information, the NRC staff concludes that the CRDM will continue to meet its design-basis and performance requirements at uprated power conditions, and is, therefore, acceptable.

3.2.5 Reactor Recirculation System

The licensee will increase power to the uprated level by operating along higher rod lines on the power/flow map with allowance for increased core flow (ICF). The cycle-specific core reload analyses will consider the full core flow range, up to 115 million lbm/hr. The licensee's evaluation of the reactor recirculation system performance at uprated power with ICF determined that the core flow can be maintained. The design pressures for the reactor recirculation control (RRC) system components include the suction, discharge, and flow control valves, as well as recirculation pumps and piping. Raising the steam pressure by 15 psi as a result of power uprate will raise the pump suction pressure by 17 psi and the pump discharge pressure by 45 psi. The licensee states that these increases in normal operating pressures are bounded by the system design pressure. Operation at uprated conditions will increase the RRC pump suction temperature by approximately 1 °F which is bounded by the system design temperature.

The pump speed and flow control valve position runback functions affected by power uprate and ELLL are changed by the amendment. The new cavitation interlock setpoint is 11 °F. The new flow control valve runback setpoint for the uprated power condition corresponds to a core flow of 48 percent of rated flow. The licensee concluded that the changes caused by power uprate and ELLL are small and are bounded by the RRC design basis. Startup testing of the RRC system is discussed in Section 3.8.4 of this SE.

The NRC staff reviewed the results of the licensee's evaluation and concludes that the existing RRC system design has sufficient margin to accommodate operation at the uprated power condition, and is therefore acceptable.

3.2.6 Reactor Coolant Piping and Components

The licensee evaluated the effects of the power uprate -- considering higher steam flow rate, temperature, and pressure for thermal expansion, dynamic loads, and fluid transient loads -- on the reactor coolant pressure boundary (RCPB) piping systems (the main steam (MS), recirculation, feedwater (FW), CRD, reactor water clean-up (RWCU), residual heat removal (RHR), high- and low-pressure ECCS, and standby liquid control (SLC) and reactor core isolation cooling (RCIC)). The licensee's evaluation was performed in accordance with requirements of the ASME Boiler and Pressure Vessel Code Section III Subsection NB-3600, 1971 Edition with Addenda through Winter 1971 (Reference 18).

The licensee stated that stresses and fatigue usage factors were calculated for the power uprate, based on equations 9 through 14 of Reference 18, for the design, normal, upset, emergency, and faulted conditions. The revised stresses resulting from the power uprate were compared with the code-allowable stresses for acceptability. The licensee concluded that the code requirements are satisfied for the piping systems evaluated and that the power uprate will not have an adverse effect on the Class 1 piping system design.

The licensee evaluated pipe supports, equipment nozzles, valves, guides, penetrations, and piping suspension devices by comparing the increased piping interface loads on the system components with the margin in the original design-basis calculation. The increased interface loads are due to thermal expansion of the piping and components from the power uprate. The licensee found sufficient margin between the original design stresses and the code limits to accommodate the stress increase for all service levels at the uprated power. The licensee also evaluated the effect of power uprate conditions on thermal and vibration displacement limits for piping, system springs, snubbers, and rigid supports and found the limits acceptable.

The NRC staff reviewed the results of the licensee's evaluation and concludes that the design of piping, components, and their supports is adequate to maintain the structural and pressure boundary integrity of the reactor coolant piping systems in power uprate conditions, and is, therefore, acceptable.

3.2.7 Main Steam Isolation Valves (MSIVs)

The licensee evaluated the MSIVs, and found them consistent with the bases and conclusions of the generic evaluation (Reference 7). Increased core flow alone does not change the conditions within the main steam lines, and thus cannot affect the MSIVs. MSIV performance is routinely monitored as required by TSs to ensure that the original licensing basis for MSIVs is preserved. This is consistent with the generic evaluation in Reference 7, and is acceptable to the staff.

3.2.8 Reactor Core Isolation Cooling System (RCIC)

The RCIC system provides core cooling when the RPV is isolated from the main condenser and the RPV pressure is greater than the maximum allowable for initiation of a low-pressure core cooling system. The licensee evaluated the RCIC system, and found it consistent with the bases and conclusions of the generic evaluation (Reference 7). The licensee committed to additional testing to address all aspects of GE SIL 377 (Reference 19). These tests will be conducted during power ascension testing for power uprate. This is acceptable to the staff.

The licensee evaluated the capability of the RCIC system to operate under power uprate conditions. The licensee found that the system response under the new load demands falls within the existing RCIC turbine and pump design margins.

The NRC staff reviewed the results of the licensee's evaluation and concludes that the RCIC system will continue to meet its design-basis and performance requirements at uprated power conditions.

3.2.9 Residual Heat Removal (RHR) System

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel and to remove decay heat from the primary coolant system after a reactor shutdown for both normal and postaccident conditions. The RHR system is designed to operate in the low-pressure coolant injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The LPCI mode of RHR is discussed in Section 3.3.2.2 of this SE. The effects of power uprate on the remaining operating modes are discussed in the following paragraphs.

3.2.9.1 Shutdown Cooling Mode

The licensee evaluated the shutdown cooling mode of the RHR system. The operational objective of this mode when used for normal shutdown is to reduce the bulk reactor temperature to 125 °F in approximately 20 hours, using two RHR loops. At the uprated power level, the decay heat is increased proportionally, which slightly increases the time required to reach the shutdown temperature. The licensee considers that this increased time has no effect on plant safety.

Regulatory Guide 1.139, "Guidance for Residual Heat Removal," (Reference 20) states that cold shutdown capability (200 °F reactor fluid temperature) should be reached within 36 hours. Final Safety Analysis Report (FSAR) Section 15.2.9 indicates that cold shutdown can be reached in a much shorter time, even considering the availability of only one RHR heat exchanger. For power uprate, the licensee performed an alternate shutdown cooling analysis based on the criteria of Regulatory Guide 1.139. The results of this analysis show that for the power uprate condition, the reactor can still be cooled to 200 °F in less than the 36-hour criterion. The NRC staff reviewed the results of the licensee's evaluation and finds this acceptable.

3.2.9.2 Suppression Pool Cooling Mode (SPCM)

The functional design basis as stated in the FSAR for the SPCM is to ensure that the pool temperature does not exceed its maximum temperature limit after a blowdown. This objective is met with power uprate, since the licensee's analysis confirms that the pool temperature will stay below its design limit at uprated conditions. Suppression pool temperature response is discussed further in Section 3.3.1.1 of this SE.

3.2.9.3 Containment Spray Cooling Mode

In the containment spray cooling mode, the RHR system provides water from the suppression pool to spray headers in the drywell and suppression chambers to reduce containment pressure and temperature during postaccident conditions. Power uprate will increase the containment spray temperature by only a few degrees. This increase will have a negligible effect on the calculated values of drywell pressure, drywell temperature, and suppression chamber pressure, since these reach peak values before containment spray actuates. The licensee evaluated the temperature increase and determined it does not affect the function or operation of the containment spray cooling mode.

The NRC staff reviewed the results of the licensee's evaluation and concludes that the RHR system will continue to meet its design-basis and performance requirements at uprated power conditions, and is, therefore, acceptable.

3.2.10 Reactor Water Cleanup (RWCU) System

The operating pressure and temperature of the RWCU system will increase slightly as a result of power uprate. The licensee evaluated the effect of these increases and has found that uprate will not adversely affect RWCU system integrity. Although increased feedwater flow to the reactor may slightly diminish the cleanup effectiveness of the RWCU system, the power uprate will not require a change in current TS limits for reactor water chemistry.

The NRC staff reviewed the results of the licensee's evaluation and concludes that the RWCU system will continue to meet its design-basis and performance requirements at uprated power conditions, and is, therefore, acceptable.

3.3 Engineered Safety Features

The NRC staff reviewed the effect of power uprate on containment system performance, the standby gas treatment system (as affected by increased iodine loading), post-LOCA combustible gas control, the control room atmosphere control system, and the emergency cooling water system. The NRC staff performed this review to verify that the uprate would not impair the ability of these systems to perform their safety functions to respond to or mitigate the effects of design-basis accidents. The NRC staff also considered the effects of the power uprate on high-energy line breaks, fire protection, and station blackout.

3.3.1 Containment System Performance

Section 5.10.2 of Reference 9 requires the power uprate applicant to show acceptability of the uprated power level for: (1) containment pressures and temperatures, (2) LOCA containment dynamic loads, and (3) safety-relief valve dynamic loads. Appendix G of Reference 9 prescribes the approach to be used by power uprate applicants for performing required plant-specific analyses. The licensee performed the necessary analyses and discussed its results in its application. The analyses results are discussed below and summarized in Table 3.1 of this SE.

Appendix G of Reference 9 states that the licensee needs to analyze short-term containment responses using the staff-approved M3CPT code. M3CPT is used to analyze the period from break initiation through initiation of pool cooling. M3CPT output is also used as input for dynamic loads analyses and equipment qualification analyses.

Appendix G of Reference 9 states that the licensee needs to analyze long-term containment heatup (suppression pool temperature) for the limiting safety analysis report events to show that pool temperatures will be within the required limits for containment design temperature, NUREG-0783 (Reference 22) local pool temperature, net positive suction head, and other considerations (e.g., pump seals, piping design temperatures). Licensees performing these analyses are to use the SHEX code and ANS 5.1-1979 decay heat assumptions consistent with Reference 21. SHEX, which is partially based on M3CPT, is a long-term code used to analyze the period from break initiation until after peak pool heatup.

3.3.1.1 Containment Pressure and Temperature Response

The WNP-2 containment consists of a drywell/vent system and a wetwell. The drywell/vent system is designed for a pressure of 45 psig and a temperature of 340 °F. The wetwell is designed for a pressure of 45 psig and a temperature of 275 °F.

(a) Long-Term Suppression Pool Temperature Response

(1) Bulk Pool Temperature

The licensee evaluated the long-term response of the bulk suppression pool temperature for 102 percent of an analyzed power level of 3629 MWt. The design-basis accident (DBA) LOCA peak temperature was calculated to be 204 °F which is bounded by the original design value of 212 °F. Current TS require that the ultimate heat sink be operable at a temperature less than or equal to 77 °F. The new analysis conservatively assumes that the service water temperature is 90 °F (the previous analysis assumed a service water temperature of 95 °F). Thus, existing TSs ensure adequate ultimate heat capacity and temperature. Section 6.4.1.1.2 of Reference 1 states that the containment cooling analysis does not assume that the post-LOCA RHR cooling capacity is increased. The licensee determined that the highest pool temperature resulting from the most severe non-LOCA event, an alternate shutdown event, would be 210 °F. This result is bounded by the design value.

The NRC staff reviewed the results of the licensee's analysis and concludes that the bulk suppression pool temperature response is acceptable for power uprate conditions.

(2) Local Pool Temperature With SRV Discharge

The higher SRV setpoints accompanying power uprate result in an approximately 3.3-percent increase in quencher mass flux and an increase in the total heat load. The licensee analyzed the local suppression pool temperature response for the limiting SRV discharge event. The resultant temperature is bounded by the 200 °F local temperature limit prescribed by NUREG-0783 (Reference 22). The NRC staff reviewed the results of the licensee's analysis and concludes that an uprated power level will not introduce the possibility of SRV discharge condensation instability.

(3) Steam Bypass Capability

The licensee evaluated steam bypass of the suppression pool due to leakage between the drywell and wetwell airspace during a LOCA to ensure that there is sufficient time for manual actuation of containment spray to prevent the containment pressure from exceeding the design limit. The allowed time for operator action is 30 minutes. The licensee found that power uprate has negligible impact on bypass effects, and that sufficient time exists for manual operator action.

The NRC staff reviewed the results of the licensee's evaluation and concludes that the steam bypass response will remain acceptable after power uprate.

(b) Containment Atmosphere Temperature Response

The licensee stated that the containment drywell design temperature of 340 °F was determined based on a bounding analysis of the superheated containment atmosphere which could result from blowdown of steam to the drywell during a LOCA. Power uprate involves an increase in the nominal reactor vessel operating pressure from 1005 to 1020 psig. The original design analysis assumed a pressure of 1040 psig, which bounds the new value. The licensee determined that the original analysis bounds the power uprate condition, and other power uprate changes (such as higher decay heat) do not significantly affect the peak containment temperature response, and thus found that the LOCA short-term peak drywell temperature response is unchanged.

The licensee's analysis indicated that in the event of a LOCA, the maximum bulk pool temperature can reach 204 °F due to power uprate. With saturation conditions, the licensee found that the maximum wetwell atmosphere temperature of 204 °F will remain below the wetwell design temperature of 275 °F.

The NRC staff reviewed the licensee's evaluation and concludes that the containment atmosphere temperature response for uprated power operation is acceptable.

(c) Short-Term Containment Pressure Response

The licensee analyzed the short-term containment response to a double-ended guillotine break of a recirculation suction line to demonstrate that the accident consequences of power uprate and ELLL operation would not result in exceeding the containment design limits. The analysis encompassed the drywell pressure, wetwell pressure, and drywell-wetwell differential pressure for the period from break initiation through blowdown for the 3702 MWt condition. On the basis of these results (see Table 3.1), the staff concludes that the short-term containment pressure response is acceptable.

The licensee proposed to modify the TS definitions, limiting conditions for operation, surveillance requirements, and bases relating to the current calculated peak containment internal pressure (P_a) of 34.7-psig to reflect a new value of P_a=38 psig. The licensee selected this value to conservatively encompass the proposed uprated power with ELLL peak pressure of 35.1 psig plus possible unforeseen future plant changes and additional uprate. This ensures that containment leak rate testing is conducted at greater than or equal to peak calculated containment pressure, to maintain the validity of the post-LOCA radiological analysis.

The NRC staff reviewed the results of the licensee's analysis and concludes that the containment pressure response following a postulated LOCA will remain acceptable after power uprate.

3.3.1.2 Containment Dynamic Loads

(a) LOCA Containment Dynamic Loads

Reference 9 specifies that the licensee should determine if the containment pressure, temperature, and vent flow conditions calculated with the M3CPT code for power uprate are bounded by the analytical or experimental conditions on which the previously analyzed LOCA dynamic loads were based. If the new conditions are within the range of conditions used to define the loads, then LOCA dynamic loads are not affected by power uprate and thus do not require further analysis.

Reference 1 states that the licensee performed analyses that verified that containment pressure, temperature, and vent flow conditions are bounded. The licensee determined that the containment response is negligibly affected by power uprate, since the loads are bounded by the test conditions used to define the original loads. The short-term analysis demonstrates that the uprate would not significantly affect parameters important for LOCA containment dynamic loads (e.g., drywell and wetwell pressure, vent flow rate, and suppression pool parameters).

The NRC staff reviewed the results of the licensee's analysis and concludes that LOCA containment dynamic loads will remain acceptable for power uprate.

(b) SRV Containment Dynamic Loads

The licensee stated that SRV containment dynamic loads include discharge line loads, pool boundary pressure loads, and drag loads on submerged structures. These loads are influenced by SRV opening setpoints, discharge line configuration, and suppression pool configuration. The SRV setpoint is the only one of these affected by power uprate. Reference 9 states that if the SRV setpoints are increased, the licensee needs to show that the SRV design loads have sufficient margin to accommodate the higher setpoints.

The licensee reanalyzed the containment dynamic loads to reflect increased SRV opening setpoints (15 psi increase in Group 1) and increased SRV setpoint tolerance (from +1%/-3% to ± 3%). The licensee discusses the effects of power uprate on the pool boundary and submerged structure loads in Reference 12. The analysis was performed using GE methods for plants having X-quenchers. The results of the reanalysis indicate that the loads remain below their design values.

The NRC staff reviewed the results of the licensee's analysis and concludes that SRV containment dynamic loads are acceptable.

(c) Subcompartment Pressurization

FSAR Section 6.2.1.2 discusses the analysis for the two primary containment subcompartments for effects of pressurization. The subcompartments are (1) the annulus between the biological shield wall and reactor vessel and (2) the drywell head. The licensee determined that these subcompartments do not need to be reanalyzed for power uprate. The licensee's conclusion is based on the phenomena of the blowdown process. Subcompartment pressurization is controlled by the initial break flow mass-energy release rates which are governed primarily by the initial conditions of the reactor coolant inventory. The mass-energy release rates are not significantly affected by power uprate.

The NRC staff agrees with the licensee that power uprate will not affect subcompartment pressurization, and concludes that subcompartment pressurization effects will remain acceptable for uprated power.

3.3.1.3 Containment Isolation

The Reference 9 methodology does not address a need for reanalysis of the isolation system. The isolation system is not adversely affected by power uprate. MSIVs are designed and installed in such a manner that increased steam flow will not adversely affect their capability to close within specified time limits. A generic analysis described in Reference 7, paragraph 4.7, concluded that existing MSIVs are acceptable for power uprate for all BWR 4/5/6 facilities subject to plant-specific confirmation that associated Class 1E components such as limit switches and solenoid valves are qualified for any increase in environmental stress resulting from power uprate. As discussed in Section 3.8.2 of this SE, the licensee has an acceptable approach to assure qualification of safety-related electrical equipment resulting from any increase in environmental stress.

3.3.1.4 Containment Combustible Gas Control

Section 2.4.5 of Reference 23 states that plant-specific submittals need to confirm the capability of the combustible gas control system, and also need to address any procedural or equipment setpoint changes that may be required to ensure adequate containment atmosphere combustible gas control.

The WNP-2 combustible gas control systems include an inerted containment and redundant postaccident hydrogen recombiners. The inerted containment serves to accommodate the early hydrogen produced as a result of metal-water reaction, whereas the recombiners serve to preclude the molecular oxygen and hydrogen buildup that could result later from radiolysis. The amount of hydrogen that is assumed to result from metal-water reaction is prescribed by Regulatory Guide 1.7 as a function of core geometry and is not affected by power uprate. The licensee confirmed that the postaccident hydrogen recombiners have sufficient capacity to accommodate the increased radiolytic oxygen production resulting from an increase in rated power level. Accordingly, the NRC staff concludes that there are no combustible gas control concerns that would preclude the proposed increase in rated power.

3.3.2 Emergency Core Cooling Systems (ECCSs)

The effect of power uprate and the increase in RPV dome pressure on each ECCS is addressed below. Also as discussed in the FSAR, compliance to the net positive suction head (NPSH) requirements of the ECCS pumps is conservatively based on a containment pressure of 0 psig and the maximum expected temperature of pumped fluids. The pumps are assumed to be operating at the maximum runout flow with the suppression pool temperature at its NPSH limit (212 °F). Assuming a LOCA occurs during operation at the uprated power, the suppression pool temperature will remain below its NPSH limit. Therefore, power uprate will not affect compliance to the ECCS pump NPSH requirements.

3.3.2.1 High-Pressure Core Spray (HPCS) System

The licensee evaluated the HPCS system and found that system design and operation is bounded by the generic evaluation for power uprate (Reference 7). The staff concludes that it provides an acceptable basis for the design and operation of the HPCS system for power uprate, and is therefore acceptable.

3.3.2.2 Low-Pressure Core Injection (LPCI) System Mode of RHR

The licensee evaluated the RHR system for LPCI operation, and found that the hardware for the low-pressure portions of the RHR is not affected by power uprate. The upper limit of the low-pressure ECCS injection setpoints will not be changed for power uprate; therefore, the low-pressure portions of these systems will not experience any higher pressures. The power uprate will not result in an increase in the licensing and design flow rates of the low pressure ECCS. In addition, the uprated power will not require an increase in the RHR system shutdown cooling mode flow rates and operating pressures.

The NRC staff reviewed the licensee's evaluation and concludes that the licensee's findings are consistent with the bases and conclusions of the generic power uprate evaluation (Reference 7), and are therefore acceptable.

3.3.2.3 Low-Pressure Core Spray (LPCS) System

The licensee evaluated the LPCS system, and found that the hardware for the LPCS is not affected by power uprate. The power uprate will not affect the upper limit of the LPCS injection setpoints; therefore, the low-pressure portions of this system will not experience any higher pressures. The power uprate will not increase the licensing and design flow rates of the low-pressure ECCS. The licensee determined that the short-term LPCS system response to a LOCA will continue to bound the long-term system response.

The NRC staff reviewed the licensee's evaluation and concludes that the licensee's findings are consistent with the bases and conclusions of the generic power uprate evaluation (Reference 7) and are, therefore, acceptable.

3.3.2.4 Automatic Depressurization Systems (ADSs)

The ADSs use SRVs to reduce reactor pressure following a small-break LOCA with HPCS failure. This function allows low-pressure coolant injection (LPCI) and core spray (CS) to flow to the vessel. The licensee evaluated the ADS initiation logic and ADS valve controls and found that they are adequate for power uprate. The licensee determined that the ECCS design requires a minimum flow capacity equivalent to five SRVs/ADS valves. SRV setpoint tolerance and out-of-service requirements are established to assure this requirement is met. ADS initiates (after a time delay) on Low Water Level 1 in conjunction with a signal that at least one LPCI or LPCS pump is running with permissive from Low Water Level 3. The licensee determined that power uprate does not affect the ability to perform these functions.

The NRC staff reviewed the licensee's evaluation and concludes that the licensee's findings are consistent with the bases and conclusions of the generic power uprate evaluation (Reference 7) and are, therefore, acceptable.

3.3.3 Emergency Core Cooling System (ECCS) Performance Evaluation

The ECCSs are designed to protect against hypothetical LOCAs caused by ruptures in primary system piping. The ECCS performance under all LOCA conditions and their analysis models need to satisfy the requirements of 10 CFR 50.46 and 10 CFR Appendix K. The licensee evaluated the Siemens Nuclear Power (SNP) 8x8 and 9x9-9x fuel, used in WNP-2, with NRC-approved methods. The results of the ECCS-LOCA evaluation are discussed in the following paragraphs.

The licensee used the NRC-approved SAFER/GESTR (S/G) methodology to assess the ECCS capability for meeting the 10 CFR 50.46 criteria. The licensee performed S/G-LOCA analysis for WNP-2 with SNP 8x8 and 9x9-9x fuel in accordance with NRC requirements. The results (Reference 11) demonstrate conformance with the ECCS acceptance criteria of 10 CFR 50.46 and Appendix K. A sufficient number of plant-specific break sizes were evaluated to establish the behavior of both the nominal and Appendix K peak cladding temperature (PCT) as a function of break size. Different single failures were also investigated in order to clearly identify the worst cases. The licensee performed the WNP-2 specific analysis using a conservatively high peak linear heat generation rate (PLHGR) and a conservatively low minimum critical power ratio (MCPR). In addition, some of the ECCS parameters were conservatively established relative to ECCS design performance. The nominal (expected) PCT is 992 °F. The statistical upper bound PCT is below 1440 °F. The licensing basis PCT for WNP-2 is 1440 °F which is well below the acceptance criterion of 10 CFR 50.46 PCT limit of 2200 °F. The analysis also conforms to the other acceptance criteria of 10 CFR 50.46. Compliance with each of the elements of 10 CFR 50.46 is documented in Table 4-2 of Reference 11.

The NRC staff reviewed the results of the licensee's analysis and concludes that WNP-2 complies with the NRC S/G-LOCA licensing analysis requirements.

The licensee also evaluated the ECCS performance for single-loop operation (SLO) using the S/G-LOCA methodology. The DBA size break is also limiting for SLO. The licensee's evaluation, using the same assumptions in the S/G-LOCA calculation with no MAPLHGR reduction, yields a calculated nominal and Appendix K PCT of 1184 °F and 1504 °F, respectively. Since the PCT was below the 10 CFR 50.46 limit of 2200 °F, the licensee determined that no MAPLHGR reduction is required for SLO. In Reference 24, the NRC staff asked the licensee to reconçile the fact that the S/G-LOCA analysis PCT results for SLO were higher than those presented for two-loop operation, and to address the concern that the licensee did not provide a statistical analysis of the upper bound PCT for this case. The licensee responded in a letter dated January 6, 1994 (Reference 3), stating that the upper bound PCT for WNP-2 is 1450 °F, which is below the 1600 °F limit. The current WNP-2 TS applies a 0.01 adder to the safety limit minimum critical power ratio (SLMCPR) when in SLO due to increased uncertainties. This is acceptable to the staff.

The licensee also evaluated the applicability of the S/G-LOCA methodology to WNP-2 which operates with Siemens Nuclear Power (SNP) 8x8 and 9x9-9x fuel. The dimensions and characteristics of the SNP fuel are similar to GE fuels. The MAPLHGR and PLHGR values used in the WNP-2 analysis are based on inputs for the SNP 8x8 and 9x9-9x fuel shown on Figures 5-2 through 5-5 in NEDC-32115P, respectively. The S/G-LOCA analysis is valid for fuel designs with comparable geometry and for MAPLHGR and PLHGR values less than or equal to those shown in the previously mentioned figures in NEDC-32115P. Since the geometry and characteristics of the SNP fuel used in WNP-2 are similar to those of a typical GE BWR plant, the S/G-LOCA methodology is considered applicable to WNP-2 with SNP fuel.

The licensee evaluated the impact of ICF, up to 115 Mlb/hr, on LOCA results at the 3629 MWt power level (corresponding to a 110-percent increase in steam flow) using S/G-LOCA methodology for WNP-2. The evaluation results for a DBA recirculation line break with the same single failure (HPCS diesel) and using the same Appendix K and nominal assumptions show a decrease in the nominal PCT when compared to the base case. This decrease in PCT for the nominal ICF case is due to (1) the better heat transfer during flow coastdown from the higher initial flow and (2) less subcooling in the downcomer which results in reduced break flow and later core uncovery.

The NRC staff reviewed the results of the licensee's analysis and concludes that the licensee's evaluation demonstrates that the ECCS systems are acceptably designed for power uprate conditions.

3.3.4 Standby Gas Treatment System (SGTS)

The SGTS is designed to ensure controlled and filtered release of particulate and halogens from primary and secondary containments to the environment during abnormal and accident situations in order to maintain thyroid doses within the 10 CFR Part 100 guidelines. The system consists of two 100-percent-capacity, parallel, redundant flow trains. Each flow train consists of a moisture separator, two electric heater banks, a prefilter, a high-efficiency particulate air (HEPA) filter, an electric strip heater, a charcoal adsorber bed, an additional electric strip heater and charcoal adsorber bed, a HEPA afterfilter, and two 100-percent-capacity exhaust fans. The capacity of the SGTS was designed to provide one secondary containment volume change per day and thereby maintain the reactor building (RB) at a slight negative pressure of 0.25-inch water gauge with respect to the outside atmosphere. The negative pressure prevents the unfiltered release of radioactive material from the RB to the environment. The licensee stated that the proposed slight uprate in power (4.9 percent) will not change the ventilation design aspects of the SGTS. The licensee concluded that the power uprate by itself will not have any adverse impact on the capability of the SGTS to meet this design objective. The NRC staff reviewed the licensee's evaluation and concludes that the power uprate by itself will not have any adverse impact on the capability of the SGTS to meet the SGTS design objectives since it does not change the ventilation design aspects of the SGTS.

The NRC staff recognizes that the iodine loading will increase marginally because of the proposed power uprate. The SGTS design uses filters that meet Regulatory Guide (RG) 1.52 criteria with respect to the design, testing, and maintenance criteria of engineered safety feature (ESF) grade filters. One of the criteria deals with the filter loading capability. The licensee stated that although the iodine loading will increase slightly, it will remain well below the original design capability of the filters. The NRC staff reviewed the licensee's evaluation and concludes that the SGTS will continue to meet the RG 1.52 criteria and is, therefore, acceptable for power uprate operation.

On these bases, the NRC staff concludes that SGTS will continue to meet its design objectives for uprated power operation.

3.3.5 Other ESF Systems

3.3.5.1 MSIV Leakage Control System

The licensee's containment analysis determined that the peak post-LOCA containment pressures will not increase beyond the original design basis due to uprated power operation. The NRC staff concludes that operation at power uprate conditions will not affect operation of the MSIV leakage control system.

3.3.5.2 Main Control Room Atmosphere Control System (CRACS)

The CRACS, containing air handling and emergency filtration units, is designed to maintain the control room envelope at a slightly positive pressure (1/8" water gauge) relative to the outside atmosphere and thus minimize unfiltered inleakage of contaminated outside air into the control room following an accident. The system accomplishes the design objective by bringing in controlled and filtered outside air to keep the control room operator doses within GDC 19 limits during an accident. The NRC staff determined that since plant operation at the uprated power level does not change the design and operational aspects of the control room emergency filtration system, there will not be a significant increase in unfiltered inleakage of contaminated outside air into the control room following an accident.

The staff recognizes that following a LOCA, iodine loading in the makeup air filters and recirculation air filters will increase marginally under uprate conditions. The licensee committed in the FSAR to design, test, and maintain the control room emergency filtration system filters in accordance with RG 1.52 guidelines. Therefore, the NRC staff concludes that the filters will continue to be valid for the control room atmosphere control system at uprate operation.

On this basis, the staff concludes that the uprated power level will have little or no impact on the CRACS meeting its design objectives.

3.4 Instrumentation and Control

Many of the TS changes proposed in the licensee's application (Reference 1) involve changes to the reactor protection system (RPS) trip and interlock setpoints. These changes are intended to maintain the same margin between the new operating conditions and the new setpoints that existed before the power uprate.

The conservative design calculations for the initial licensing of WNP-2 resulted in setpoints that provided excess reactor coolant flow capacity and corresponding margins in the power conversion system. For WNP-2, these margins (e.g., 5 percent rated steam flow) result in the capability to increase the core operating power level by approximately 4.9 percent. This SE is limited to the setpoint changes for the identified instrumentation and is predicated on the assumption that the analytical limits used by the licensee are based on application of approved design codes.

A review of the licensee's submittal indicates that plant-specific calculations were performed using methods recommended by the Instrument Society of America as outlined in GE Topical Report NEDC-31336-P (Reference 25).

The licensee proposed the following setpoint changes:

(a) APRM Flow Biased Simulated Thermal Power

(1) Flow Biased

Change trip from 0.66W + 51% to 0.58W + 59%.

Change Allowable Value from 0.66W + 54% to 0.58W + 62%.

(2) Flow Clamped

No change in trip setpoint.

Change Allowable Value from ≤115.5% to ≤114.9%.

(b) Reactor Vessel Steam Dome Pressure High

Change trip from 1037 psig to 1060 psig.

Change Allowable Value from 1057 psig to 1074 psig.

(c) Main Steam High Flow

Change trip from 105.5 psid to 115.6 psid.

Change Allowable Value from 108 psid to 124.6 psid.

(d) Rod Block Monitor - Flow-Biased-Upscale

Change trip from 0.66W + 40% to 0.58W + 48%.

Change Allowable Value from 0.66W + 43% to 0.58W + 51%.

(e) APRM Rod Block - Flow-Biased Neutron Flux Upscale

Change trip from 0.66W + 42% to 0.58W + 50.0%.

Change Allowable Value from 0.66W + 45% to 0.58W + 53.0%.

(f) Turbine Stop Valve and Turbine Control Valve Fast Closure Scram Bypass

The turbine first-stage pressure setpoint was changed to reflect the expected pressure at the new 30-percent power point.

The licensee's submittal dated July 9, 1993 (Reference 1) did not describe the methodology used for instrument setpoint calculations. Therefore, in a letter dated November 9, 1993 (Reference 30), the NRC staff requested additional information regarding the instrument setpoint methodology. The licensee, in a letter dated February 2, 1994 (Reference 4), responded to the staff's request and confirmed that it used GE licensing Topical Report NEDC-31336-P (Reference 25) for calculating instrument setpoints. The NRC staff previously reviewed this topical report and accepted it with some minor exceptions. The NRC staff is reviewing these exceptions and will resolve them generically. They do not affect the staff's evaluation of the proposed WNP-2 change.

The proposed setpoint changes are intended to maintain the existing margins between operating conditions and reactor trip setpoints. Thus, margins to the new safety limits will remain the same as the current margins. These new setpoints also do not significantly increase the likelihood of a false trip nor failure to trip upon demand.

The NRC staff concludes that the licensee's instrument setpoint methodology and the resulting setpoint changes incorporated in the TSs for power uprate are consistent with the WNP-2 licensing basis and are, therefore, acceptable.

3.5 Auxiliary Systems

3.5.1 Spent Fuel Pool Cooling and Cleanup System (FPCCS)

The FPCCS is designed to remove the decay heat generated by the stored spent fuel assemblies. The system consists of fuel pool cooling pumps, heat exchangers, skimmer surge tanks, filter demineralizers, associated piping, valves and instrumentation. The system is designed to transfer the decay heat generated by the spent fuel to the reactor building closed cooling water system, and can be cooled by the service water system during abnormal operation.

Spent fuel discharged to the pool following operation at the uprated power level will increase the heat load on the FPCCS. The licensee evaluated spent fuel pool heat loads and radiological consequences for plant operations at the uprated power level. The results of the evaluation indicate that the fuel pool temperature can be maintained below the 150 °F FSAR limit with two fuel pool cooling heat exchangers in service under the maximum expected normal heat load using uprated power. The licensee analyzed the failure of a single cooling train under the same heat loading conditions and determined that the 155 °F FSAR limit for a single failure would not be exceeded.

The FPCCS is capable of maintaining pool temperatures less than 145 °F for the emergency full core offload with the fuel pool at maximum capacity. Supplemental RHR cooling is required immediately after the full core offload. The FSAR limit for this scenario is 175 °F.

The licensee verified that the spent fuel racks are designed to withstand the higher fuel temperatures expected from the power uprate. The licensee also determined that the capacity of the standby service water system is adequate to cool the spent fuel pool with the anticipated increase in heat load.

The licensee conducted its evaluation by reviewing the original design requirements and bases for the FPCCS, the current plant operating conditions, and the assumptions made for the proposed power level increase. Findings indicate that the proposed power uprate will have a minimal effect on the FPCCS. Although normal radiation levels around the pool are expected to increase, primarily during fuel handling operations, the licensee considers this increase acceptable and does not anticipate a significant increase in operational doses to personnel or equipment.

The NRC staff reviewed the results of the licensee's evaluation and concludes that operation at uprated power will not prevent the FPCCS from performing its design function. In addition to the licensee's evaluations, the NRC staff identified an issue associated with the adequacy of spent fuel pool cooling in NRC Information Notice (IN) 93-83, "Potential Loss of Spent Fuel Pool Cooling Following a Loss of Coolant Accident (LOCA)," October 7, 1993, and in a 10 CFR Part 21 notification dated November 27, 1992. The staff is evaluating this issue, as well as broader issues associated with spent fuel storage safety, as part of the NRC generic issue evaluation process. If the generic review concludes that additional requirements in the area of spent fuel pool safety are warranted, the staff will address those requirements to the licensee under separate cover.

Based on its review of the licensee's evaluation and consideration of potential generic issues identified in NRC IN 93-83, the NRC staff concludes that operation at uprated power will have little or no impact on the spent fuel pool cooling system operation at WNP-2.

3.5.2 Cooling Water Systems

The licensee evaluated the effect of power uprate on the various plant water systems, including safety-related and non-safety-related service water systems, the closed-loop cooling water system, and the plant ultimate heat sink. The licensee's evaluation took into account the increased heat loads, temperature, pressures, and flow rates. The following sections discuss the specific evaluations performed.

3.5.2.1 Safety-Related Loads

The safety-related heat loads are rejected to one of two safety-related service water systems. These systems are the emergency equipment service water system and the residual heat removal service water system. All heat removed from these systems is then rejected to the ultimate heat sink (UHS). The NRC staff's evaluation of the effects of uprated power level operation on each of these subsystems appears below.

(a) Emergency Equipment Service Water System (EESWS)

The licensee evaluated the EESWS for its ability to provide cooling water to ECCS and ESF components, heating, ventilation, and air conditioning (HVAC) systems, and diesel generator coolers. The system provides water from the UHS spray pond through the various equipment coolers and returns it to the pond through a spray network. The licensee found that the loads on the EESWS were not power dependent and thus remain unchanged for LOCA conditions following uprated operation.

The NRC staff reviewed the licensee's evaluation and concludes that the proposed power uprate will have little or no effect on the EESWS meeting its design objectives.

(b) Residual Heat Removal (RHR) Service Water System

The RHR service water system provides safety-related cooling water to the RHR system under normal and postaccident conditions. The system provides water from the UHS (spray pond) through the RHR heat exchangers and returns it to the pond via a spray network.

The licensee evaluated the effects of power uprate on the RHR service water system, and found that the system's post-LOCA RHR cooling capacity remains unchanged.

The NRC staff reviewed the results of the licensee's evaluation and concludes that power uprate will not have a significant effect on the cooling requirements for the RHR service water system.

3.5.2.2 Non-Safety-Related Loads

The licensee stated that the increase in plant service water system (PSWS) heat loads should be proportional to the uprated power level operation. The licensee evaluated the PSWS to ensure that it is capable of supplying sufficient cooling water to remove the additional heat load from the proposed power uprate without making modifications to the existing system.

Since the PSWS does not perform any safety function, the NRC staff did not review the effect of the uprated power level operation on the PSWS design and performance.

3.5.2.3 Main Condenser/Circulating Water System/Normal Heat Sink

The main condenser, circulating water system, and normal heat sink condense steam in the condenser and reject heat to the circulating water system. This maintains a vacuum in the condenser to provide for efficient turbine performance by maintaining condenser backpressure. The licensee evaluated the performance of the main condenser and found that the condenser, circulating water system, and normal heat sink were adequate for uprated power operations.

Since the main condenser, circulating water system, and normal heat sink do not perform any safety function, the staff did not review the effect of the uprated power level operation on the design and performance of these systems.

3.5.2.4 Reactor Building Closed Cooling Water (RBCCW) System

The RBCCW system is designed to remove heat from the non-safety-related equipment located in the reactor building during normal plant operations and provide a barrier between systems carrying radioactive fluids and the non-radioactive service water system. The licensee evaluated the RBCCW system and found that the increase in heat load due to the power uprate will not have a significant impact on the capability of the RBCCW system to perform its design function. The licensee determined that the maximum heat load to the RBCCW system including the additional heat load resulting from the power uprate will be 46.5 MBtu/hr, which is below the 50.0 MBtu/hr capacity of the RBCCW system. The licensee stated that it will modify flow to equipment affected by the power uprate and cooled by the RBCCW system as required to support the power uprate.

The NRC staff reviewed the results of the licensee's analysis and concludes that the effect of uprated power operations on the RBCCW system is negligible and the system has sufficient heat removal capability to accommodate operations at the increased power level.

3.5.2.5 Ultimate Heat Sink (UHS) System

The UHS is a source of safety-related cooling water for the standby service water system during normal shutdown and accident conditions. The UHS consists of a spray pond system capable of performing its safety function without offsite power available. The UHS provides a cooling capability for a period of 30 days without any outside makeup water.

The licensee evaluated the UHS at uprated power conditions and found that the post-LOCA UHS water temperature will decrease from the temperature predicted in the licensee's current FSAR. The NRC staff attributed this decrease to the licensee's use of a more refined decay heat model (ANSI/ANS 5.1-1979) that assumes a lower integrated heat addition. This model was not available when the WNP-2 FSAR was developed, and the NRC staff has not accepted the model for use at WNP-2. The NRC staff requested (Reference 26) that the licensee address this issue. The licensee responded to the request for additional information (RAI) (Reference 5) and discussed the results of an analysis using the decay heat model used in the FSAR. The licensee concluded that even though the proposed power uprate will increase the evaporation rate of the spray pond system from 1 percent to 4 percent, the minimum required inventory of the UHS system will not be affected. In conducting both analyses, the licensee conservatively assumed a 5-percent increase to all major components which comprise the total heat load to the UHS.

The licensee found that the UHS system will continue to provide a sufficient quantity of water at a temperature less than 88.6 °F (FSAR peak spray pond design temperature) following a DBA-LOCA, and no changes to the TS for the UHS are required.

The NRC staff reviewed the results of the licensee's evaluation and concludes that the UHS system design is adequate for the uprated power operation.

3.5.3 Standby Liquid Control System (SLCS)

The licensee evaluated the SLCS and found that the ability of the SLCS to achieve and maintain safe shutdown is not directly affected by core thermal power; rather, it is a function of the amount of excess reactivity present in the core; and as such, is dependent upon fuel-loading techniques and uranium enrichment. The SLCS is designed to inject at a maximum pressure equal to that of the lowest safety/relief valve setpoint. The SLCS pumps are positive displacement pumps, and the small (15 psig) increase in the lowest safety/relief valve setting as a result of power uprate will not impair the performance of the pumps.

The NRC staff reviewed the licensee's analysis and concludes that the ability of the SLCS to inject water to the reactor will not be impaired by power uprate.

In the future, however, the licensee may want to increase fuel enrichments in order to meet fuel energy requirements for longer fuel cycles. The increased excess reactivity associated with this increase in fuel enrichment will affect the reactivity requirements of the SLCS. The licensee will evaluate the SLCS requirements for future operating cycles on a cycle-specific basis.

3.5.4 Power-Dependent Heating, Ventilation, and Air Conditioning (HVAC) Systems

The licensee evaluated the impact of higher process fluid temperatures in piping for all HVAC systems, including units in the reactor building, turbine building, drywell, steam tunnel, radwaste building, and control room area.

The licensee stated that the uprated heat loads in the radwaste building, control room area, drywell, steam tunnel, and reactor building have minimal impact on maintaining the design environmental temperature parameters since the uprated parameters are within the scope of the original SE performed for WNP-2. The licensee stated that operations at the uprated power level would cause some areas in the turbine building to exceed their design temperatures. The licensee evaluated the effect of the elevated temperatures and found that there is no safety equipment in these areas for which the equipment design capability is exceeded.

The NRC staff reviewed the licensee's evaluation and concludes that operations at the uprated power level will have minimal impact on the plant HVAC systems and is, therefore, acceptable.

3.5.5 Fire Protection Systems

The licensee stated that operation of the plant at the uprated power level will not affect the fire suppression or fire detection systems. The licensee stated that there are no physical plant configuration or combustible load changes resulting from the uprate, and that the safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change.

The licensee's plan for addressing unresolved issues associated with certain post-fire safe-shutdown operator actions are documented in the licensee's letter of July 15, 1994. The licensee's proposed final resolution of these issues is documented in the licensee's letter dated January 25, 1995. The NRC staff reviewed the issues and found that they do not adversely impact the proposed power uprate. The NRC staff will complete its review of the post-fire safe-shutdown operator actions as a separate licensing issue.

The NRC staff agrees that power uprate will not affect fire detection and suppression systems and, therefore, concludes that the fire protection program is not affected by power uprate.

3.5.6 Power Conversion Systems

The steam and power conversion systems and associated components (e.g., the turbine/generator, condenser and steam jet air ejectors, turbine steam bypass, feedwater and condensate systems, etc.) were originally designed to use 105 percent of the rated power available from the NSSS. Since the requested uprated values are less than or equal to the values used in existing analyses, the NRC staff concludes that the operation at uprated power should not have a significant impact on the power conversion systems.

3.6 Radwaste Systems and Radiation Sources

3.6.1 Liquid Waste Management System

The liquid waste processing system collects, monitors, processes, stores, and returns processed radioactive waste to the plant for reuse or for discharge. The single largest source of liquid waste is from the backwash of the condensate filter demineralizers. With the power uprate, the average time between backwash/precoat will be reduced slightly. Also, the activated corrosion products in liquid wastes are expected to increase proportionally to the power uprate. However, the total volume of processed waste is not expected to increase appreciably since the only significant increase in processed waste comes from the more frequent backwashes of the condensate filter demineralizers. The licensee evaluated plant operating effluent reports and the slight increase expected from the power uprate, and found that the requirements of 10 CFR Part 20 and Appendix I to 10 CFR Part 50 will continue to be met.

Based on its review of available plant data and experience with other power uprates, the NRC staff concludes that there will not be a significant adverse effect on liquid effluents from the proposed power uprate.

3.6.2 Gaseous Waste Management Systems

The gaseous waste management systems collect, control, process, store, and dispose of gaseous radioactive waste generated during normal operation and abnormal operational occurrences. The gaseous waste management systems include the offgas system, standby gas treatment system (SGTS), and various building ventilation systems. The licensee states that the systems are designed to conform to the requirements of 10 CFR Part 20 and Appendix I to 10 CFR Part 50.

Building ventilation systems control airborne radioactive gases by using combinations of such devices as HEPA and charcoal filters, and radiation monitors that signal automatic isolation dampers or trip supply and/or exhaust fans, or by maintaining negative air pressure, where required, to limit the migration of gases. The licensee states that activity of airborne effluents released through building vents is not expected to increase significantly with power uprate because the amount of fission products released into the coolant depends on the number and nature of the fuel rod defects, and is approximately linear with respect to core thermal power. The release of gaseous effluents is an administratively controlled variable, and is not a function of core power.

Based on its review of available plant data and experience with other power uprates, the NRC staff concludes that the proposed power uprate will not have a significant adverse effect on airborne effluents.

3.6.3 Radiation Sources in the Core and Coolant

During reactor operation, the coolant passing through the core region becomes radioactive as a result of nuclear reactions. Radioactive materials in the reactor core are produced in direct proportion to the fission rate. Thus, the levels of radioactive materials (for both fission products and activation products) produced are expected to increase by a maximum of 4.9 percent. The licensee evaluated the effect of the power uprate on coolant activation products, activated corrosion products, and fission products and found that they are expected to be approximately equal to current measured data which is within the design basis of the plant.

Based on its review of available plant data and experience with other power uprates, the NRC staff concludes that the power uprate will not have a significant adverse effect on radiation sources in either the core or reactor coolant.

3.6.4 Radiation Levels

The licensee evaluated the effect of power uprate on radiation levels in the WNP-2 facility during normal and abnormal operation as well as from postulated accident conditions. The licensee found that radiation levels from both normal and accident conditions may increase slightly. However, any such increases would be slight and would be bounded by conservatism in the original plant design and analysis. The licensee also found that individual worker exposures will be maintained within acceptable limits by the site "as low as reasonably achievable (ALARA)" program, which controls access to radiation areas.

The licensee determined that operation at the uprated power level will not significantly affect offsite doses associated with normal operation since these releases are administratively controlled and are expected to remain below the levels in Appendix I to 10 CFR Part 50.

Based on its review of available plant data and experience with other power uprates, the NRC staff concludes that no significant adverse effect on radiation levels will result (either onsite or offsite) from the planned power uprate conditions.

3.7 <u>Reactor Safety Performance Evaluation</u>

3.7.1 Reactor Transients

The licensee evaluated the limiting plant transients. The licensee evaluated disturbances of the plant caused by a malfunction, a single failure of equipment, or personnel error according to the type of initiating event. The licensee will use its NRC-approved licensing analysis methodology to evaluate the effect of the limiting reactor transients. The licensee identified the limiting events for WNP-2 in Reference 1, which are the same as those in the generic report on power uprate. The generic guidelines also identified the analytical methods, the operating conditions that are to be assumed, and the criteria that are to be applied. Representative changes in core critical power ratios (CPRs) for analyzed transients were given; however, specific core operating limits will be supplied for each specific fuel cycle in the core operating limits report (COLR). The licensee discussed the power uprate with ELLL operation for a representative core using the GEMINI transient analysis methods listed in the generic report.

The licensee will confirm the acceptability of the safety limit minimum critical power ratio (SLMCPR) for each operating fuel cycle at the time of the reload analysis using NRC-approved methodology.

The licensee evaluated limiting transients for each category to determine their sensitivity to core flow, feedwater temperature, and cycle exposure. The licensee used the results from these analyses to develop the licensing basis for transient analyses at uprated power with ELLL operation. The licensee discussed the limiting transient results in Table 9-2 of Reference 1. The licensee will perform cycle-specific analyses at each reload and will provide the results to the NRC in the COLR. This approach is acceptable to the staff.

3.7.2 Design-Basis Accidents

The licensee evaluated plant-specific radiological consequences at uprated conditions for loss-of-coolant-accident (LOCA), the fuel handling accident (FHA), and the control rod drop accident (CRDA). The licensee calculated whole-body and thyroid doses at the exclusion area boundary (EAB) and low population zone (LPZ) for all accidents, and in the main control room for the LOCA. The licensee found that the plant-specific results for power uprate will remain well below the reference values of 10 CFR Part 100. The NRC staff compares doses resulting from the accidents analyzed to the applicable guideline in Table 3.2 of this SE.

On this basis, the NRC staff concludes that the analyzed consequences of postulated accidents remain within NRC staff acceptance criteria and are, therefore, acceptable.

3.7.3 Anticipated Transients Without Scram (ATWS)

GE performed a generic evaluation for the ATWS events. The evaluation is discussed in Section 3.7 of Supplement 2 of Reference 7. This evaluation concludes that the results of an ATWS event are acceptable for the fuel and the reactor pressure vessel (RPV), and that the containment response is also acceptable for a power uprate of 4.3 percent. The WNP-2 power increase is 4.9 percent, which is 0.6 percent above the generic evaluation. The licensee performed a WNP-2-specific ATWS analysis for a 10-percent power uprate for the limiting transients to provide assurance that the generic results will be met for WNP-2. The licensee found the results of this analysis for the ATWS event acceptable for the fuel and RPV, and the containment response is also acceptable for a 4.9-percent power uprate. GE performed a cycle-specific analysis with the major parameters and characteristics of the SNP fuel; the results were bounded by the generic ATWS analysis. The licensee found that the plant's response to an ATWS event is acceptable.

The NRC staff determined that the licensee's results were bounded by the generic analysis and therefore concludes that the plant design for response to an ATWS is acceptable.

3.7.4 Station Blackout (SBO)

The licensee evaluated the WNP-2 plant responses to a postulated SBO at a steam flow increase of 110 percent for power uprate. This corresponds to an increase of reactor thermal power to 3629 MWt from 3323 MWt. The WNP-2 response to a postulated SBO uses the RCIC and HPCS for core cooling. The licensee performed a coping evaluation to demonstrate plant response for 4 hours using HPCS with backup provided by the RCIC system. The RCIC system is the preferred source for initial operation. The licensee found that no changes to the systems or equipment used to respond to a SBO are necessary due to power uprate. The suppression pool temperature remained within design conditions. The licensee found that all equipment that takes suction from the suppression pool will operate acceptably when power is restored.

The individual considerations evaluated for power uprate included the following: the regulatory basis; the event scenario; condensate inventory and reactor coolant inventory; station battery load; compressed air supply; and loss of ventilation to the control room, reactor protection system rooms and switchgear rooms, HPCS pump and auxiliary rooms, RCIC room, containment, suppression pool, and spent fuel pool.

The NRC staff reviewed the results of the licensee's evaluation and concludes that the power uprate will have little or no effect on the plant's coping capabilities for an SBO event, and that no changes are needed to the required coping time or to systems and equipment used to respond to an SBO event. Therefore, the NRC staff concludes that the plant design for response to an SBO is acceptable.

3.8 Additional Aspects of Power Uprate

3.8.1 High-Energy Line Breaks (HELBs)

The licensee will need to slightly increase the RPV dome operating pressure to supply more steam to the main turbine. This slight increase in the operating pressure and temperature resulting from plant operations at the power uprate will cause a small increase in the mass and energy release rates following an HELB outside the primary containment. This results in a small increase in the subcompartment pressure and temperature profiles and a negligible change in the humidity profile. The licensee evaluated the HELB for the subject piping systems and found that there is no change in the postulated break locations from the uprated conditions. The licensee evaluated HELBs for the systems with the most limiting environment qualification profiles (temperature, pressure, and humidity) and found that for the RCIC line break, the original analysis envelopes the uprated conditions, and for the RWCU line break, the effects of increased mass and energy release are small and of a short duration, and have minimal impact on equipment qualification.

The licensee evaluated the effects of power uprate on plant systems and components due to moderate-energy line breaks (MELBs). The licensee determined that the original MELB analysis bounds the conditions resulting from the proposed power uprate.

The licensee evaluated the calculations supporting the disposition of potential targets of pipe whip and steam jet impingement from the postulated HELBs and determined that they are adequate for the safe-shutdown effects in the uprated power condition. The licensee determined that existing pipe whip restraints and jet impingement shields and their supporting structures are adequate for the power uprate.

The NRC staff reviewed the results of the licensee's evaluation and concludes that the analysis for HELBs is acceptable for the proposed power uprate.

3.8.2 Equipment Qualification (EQ)

3.8.2.1 Qualification of Electrical Equipment

The licensee will evaluate safety-related electrical equipment inside and outside containment to ensure qualification for the normal and accident conditions expected in the areas in which the equipment is located.

For equipment located inside the containment, the licensee determined that normal and accident design conditions for temperature, pressure, and humidity in the affected areas are unchanged for power uprate. However, normal and accident radiation levels will increase proportionally to the uprated power.

For equipment located outside the containment, qualification is based on accident temperature, pressure, and humidity resulting from an MSLB, or other HELB. The licensee determined that normal and accident temperature, pressure, and humidity conditions in the affected areas do not change with the uprated power level. However, the maximum accident radiation levels used for qualification of equipment outside containment is based on a DBA/LOCA and will increase proportionally to the uprated power.

If the reevaluation of EQ inside and outside containment identifies equipment that has the potential to be affected by the power uprate, the licensee will resolve the qualification of the equipment by performing refined radiation calculations (location specific) or by reducing the qualified life of the equipment.

The NRC staff reviewed the licensee's evaluation and commitment to resolve qualification of safety-related electrical equipment, and concludes that the licensee's approach to the qualification of safety-related electrical equipment for power uprate is acceptable.

3.8.2.2 EQ of Mechanical Equipment With Non-Metallic Components

The licensee will reevaluate safety-related mechanical equipment with non-metallic components to identify equipment affected by the uprated radiation conditions. If equipment is found to be affected by uprated radiation conditions, the qualification of the equipment will be resolved by performing location-specific radiation calculations or by reducing the qualified life of the equipment.

As stated in Section 10.2.1 of Reference 1, the normal and accident temperature, pressure, and humidity inside and outside containment are unchanged by the power uprate, and thus do not thermally affect the non-metallic components of qualified mechanical equipment. The licensee stated in a phone conversation on July 27, 1994, that these plant conditions also apply to Section 10.2.2 of Reference 1.

The NRC staff reviewed the licensee's evaluation and commitment to resolve qualification of safety-related mechanical equipment, and concludes that the licensee's approach to the qualification of safety-related mechanical equipment with non-metallic components for power uprate is acceptable.

3.8.3 Balance-of-Plant Piping Systems

The licensee evaluated the balance-of-plant (BOP) piping systems by comparing the original design-basis conditions with those for the proposed uprated conditions. The licensee also performed stress analyses in accordance with requirements of the code and the code addenda of record under the power uprate conditions. For the limiting BOP piping systems, ratios of the maximum calculated stresses at the uprated power conditions to the corresponding code-allowable stresses are summarized in Tables 2 and 3 of Reference 6. The results show that the calculated piping stresses are within the code-allowable limits. The licensee did not identify any new postulated pipe break locations in the systems evaluated.

The licensee evaluated the supports of the BOP piping systems by reviewing the increase in the pipe support loadings and stresses due to increase in the pressure, temperature, and flow rate in the affected piping systems. The licensee determined that there is sufficient margin between the original design stresses and the code limits of pipe supports to accommodate the stress increase due to the power uprate.

The NRC staff reviewed the results of the licensee's evaluation and concludes that the BOP systems will operate at the proposed power uprate conditions without adverse effects on the piping system and pipe supports.

3.8.4 Startup Testing Program

The licensee committed to a startup test program as described in Reference 9. The startup test program includes system testing of such process control systems as the feedwater flow control system. The licensee will collect steady-state operational data from 90-percent power up to the previously rated thermal power so that predicted equipment performance characteristics can be verified. The licensee will conduct the startup testing program in accordance with its procedures. The licensee also committed to include acceptance testing of the RCIC system in the startup test program. The staff finds that the licensee's approach conforms to the test guidelines in Reference 9, and is acceptable.

The NRC staff considers a testing program on the reactor recirculation (RRC) system necessary during startup subsequent to power uprate to demonstrate flow control over the entire flow range to enable a complete calibration of the flow control instrumentation, including signals to the process computer. As stated in Reference 9, these tests should also ensure that no undue vibration occurs at uprate or ELLL conditions. In response to a request for additional information dated October 26, 1993 (Reference 24), the licensee committed, in a letter dated January 6, 1994 (Reference 3), to do startup testing after making the power uprate modifications, and prior to operating in the ELLL region. The NRC staff finds this commitment conforms to the test guidelines in Reference 9, and is acceptable.

3.8.5 Equipment Seismic and Dynamic Qualification

The licensee evaluated plant equipment for seismic and dynamic qualification. The licensee found that

(a) Seismic loads are unchanged by power uprate.

(b) The original LOCA dynamic loads (pool swell, condensation oscillation (CO) and chugging, annulus pressurization, and jet impingement) are not significantly affected by the power uprate conditions as discussed in Section 4.1.2 of Reference 10.

(c) No new pipe break locations resulted from the uprated conditions.

(d) The increased SRV loads due to the increase of SRV opening setpoint pressure and the setpoint tolerance are within the margins in the plant load definition for these loads.

(e) The increased temperature, pressure, and flow conditions for power uprate on various safety-related mechanical component internals do not exceed the equipment design criteria.

The NRC staff concludes that the above information confirms that the original seismic and dynamic qualification of the safety-related mechanical and electrical equipment is not affected by the power uprate conditions and is, therefore, acceptable.

3.9 Evaluation of Effect of Power Uprate on Responses to Generic Communications

In Reference 7, GE submitted an assessment of the effect of power uprate on licensee responses to generic NRC and industry communications. GE reviewed both NRC and industry communications to determine if parameter changes associated with power uprate could affect previously made licensee commitments or earlier responses. A large number of documents were reviewed (more than 3000 items); GE nóted that only a small number of these would be affected by power uprate. The list of affected topics was then divided into those that could be bounded generically by GE, and those that would require plant-specific reevaluation. The NRC staff audited the GE assessment in December 1991 and approved the assessment in Reference 23. In addition to assessing those items requiring a plant-specific reevaluation, the licensee is also reviewing the potential effects of power uprate on internal commitments and procedures. The licensee determined that the plant-specific issues were either acceptable for power uprate, or have been revised to reflect the uprated conditions. The licensee committed to resolve any changes to commitments to include the uprated conditions. The NRC staff may audit these activities after plant startup following implementation of power uprate modifications. The NRC staff finds this approach acceptable.

3.10 Additional Changes Requested by the Supply System

3.10.1 Safety/Relief Valve (SRV) Setpoint Tolerance and Out-of-Service Analysis

The SRVs provide three main protection functions: (1) overpressure relief operation (power relief mode), in which the valves open automatically to limit a reactor vessel pressure rise; (2) overpressure safety operation (spring safety mode) to prevent reactor vessel overpressurization; and (3) depressurization operation; the automatic depressurization system (ADS) valves open automatically as part of the ECCS, for events involving small breaks in the reactor pressure boundary. GE submitted a generic topical report (Reference 28) for NRC review that supports changes to the current requirements for SRV setpoint tolerance changes. The staff prepared a safety evaluation report (SER) that approved relaxing the setpoint tolerance limit from ±1% to ±3%, when plant-specific analyses are submitted to support these changes (Reference 27). The NRC SER specified that each licensee implementing these changes must prepare certain plant-specific analyses to include the following:

(a) Transient analysis of all abnormal operating occurrences (AOOs) listed in Reference 28 using the ±3-percent setpoint tolerance for the SRVs and using staff-approved methodology.

(b) Analysis of the design overpressure event using the ±3-percent tolerance limit to confirm that the vessel peak pressure does not exceed the ASME Boiler and Pressure Vessel Code upset limit.

(c) The plant-specific analyses in Items (a) and (b) above should ensure that the number of SRVs analyzed corresponds to the number required to be operable in the TSs.

(d) Reevaluation of the performance of high-pressure systems (pump capacity, discharge pressure, etc.) motor-operated valves, and vessel instrumentation and associated piping must be completed, considering the ±3-percent limit.

(e) Evaluation of the ±3-percent tolerance on any plant operating mode such as increased core flow, extended operating domain (ELLL), and power uprate must be completed.

(f) Evaluation of the effect of the ±3%-percent tolerance limit on the containment response during LOCAs and the hydrodynamic loads on the SRV discharge lines must be completed.

The licensee submitted an evaluation (Reference 12) to support an increase in SRV setpoint tolerance and number of SRVs allowed out of service (OOS) for WNP-2. The report was in support of modifying the current in-service opening pressure setpoint tolerance from +1%/-3% to ±3%, and allowance for up to two ADS valves OOS. The licensee performed its analysis assuming a thermal power level of 3629 MWt, corresponding to 104.1 percent of the uprated power level of 3486 MWt. The analysis addresses a core flow operating range from 108.5 Mlbm/hr to 115 Mlbm/hr at the thermal power of 3486 MWt and an operating pressure of 1035 psia which corresponds to the WNP-2 power uprate with ELLL operation. A setpoint tolerance of +3 percent above the nominal SRV setting was assumed in the analysis.

The SRV setpoint tolerance and OOS analysis integrates the analysis supporting overpressure protection, containment response, SRV load definition, emergency core cooling system (ECCS) loss-of-coolant accident (LOCA) analysis events, abnormal operating occurrences (AOOs), anticipated transients without scram (ATWS), high-pressure core spray/reactor core isolation cooling (HPCS/RCIC) performance, standby liquid control system (SLCS) performance, common mode failure concerns, and Appendix R events.

The licensee's analysis, assuming MSIV closure with indirect flux scram, with six SRVs OOS, and an initial operating pressure of 1050 psia, resulted in a peak reactor vessel bottom head pressure of 1335 psig. This complies with the ASME Code allowable value of 1375 psig, and is acceptable.

The most limiting thermal transient is the load rejection with bypass failure event coincident with end-of-cycle (EOC) recirculation pump trip (RPT) OOS. The licensee analyzed this event to determine the operating limit MCPR to ensure safe plant operation. The licensee found that the peak MCPR occurs before the SRVs open at the setpoint tolerance, and the change, therefore, has no impact on thermal limits.

The ECCS-LOCA analysis assumed that the SRVs would open at the +3 percent above the nominal setpoints with four SRVs OOS. The break spectrum results show that, for the large breaks, the reactor vessel depressurizes through the break, while smaller breaks require the ADS for depressurization. The small-break analysis assumed two ADS valves OOS. The results shown in Table 4-2 of Reference 7 show that the peak clad temperature (PCT) for small breaks conforms to the 10 CFR 50.46 limit with various single failures assumed.

The HPCS and RCIC performance were evaluated for loss-of-feedwater (LOFW) events. The HPCS was evaluated assuming (1) no RCIC is available, (2) low-pressure ECCS pumps are available, (3) six SRVs OOS with relief mode 30 psi above the new nominal setpoints, (4) SRVs close at 50 psi below the opening pressure, and (5) the reactor water level initially at Level 3. The results show that the water level remains above the active fuel, which is acceptable. The RCIC was evaluated with the same assumptions as the HPCS analysis, except the HPCS and low-pressure ECCS pumps were assumed OOS. The licensee's results show no core uncovery, which is acceptable. Even though six SRVs were assumed OOS, there are other limiting concerns that will allow fewer OOS SRVs. These limiting concerns will govern the number of SRVs that will be allowed OOS.

The licensee evaluated the SLCS performance. The operating pressure range for SLCS is based on the lowest setpoint SRV available in the relief mode. The ability of the SLCS pumps to inject their design flow at higher pressures is not affected because these pumps are positive displacement-type pumps and are designed to provide constant flow regardless of system pressure. The licensee stated that the electric motors to drive these pumps have sufficient horsepower margin to meet the pump power requirements.

The effect of SRVs OOS was also evaluated for the limiting ATWS event. The reactor is eventually shut down by the SLCS during an ATWS. Initially the power is reduced by the recirculation pump trip (RPT) signal. The vessel experiences maximum pressure during the initial portion of the event. After the RPT and subsequent actuation of the SRVs, the increasing pressure transient is terminated. With four SRVs OOS, the peak pressure was calculated to be 1467 psig, which is below the ASME Code Service Level C value of 1500 psig. The licensee found that four SRVs OOS does not violate the overpressurization criterion for an ATWS event.

The licensee evaluated the common mode failure aspects of the number of SRVs allowed to be OOS. The condition with a large number of SRVs inoperable may be indicative of a common mode failure mechanism. This is the limiting concern that determines how many SRVs will be allowed to be OOS.

The licensee also evaluated the increased SRV setpoint tolerance of ±3 percent. In order to ensure that the SRV setpoints do not drift beyond the allowable ±3-percent tolerance range, the staff requested in Reference 29 that the licensee place a restriction in the proposed technical specification (TS) for the SRVs that the valves be reset to ±1-percent before returning them to service. Without this restriction, the NRC staff expected that any valve found to have significantly drifted, yet not outside the ±3-percent tolerance limit, would continue to drift, possibly outside the ±3-percent range.

Resetting the valves would provide assurance that the SRVs remain operable to accomplish their analyzed safety functions. However, the licensee stated in Reference 6 that the imposition of this requirement would result in a significant hardship. Specifically, the licensee stated that this requirement would cause multiple startups and shutdowns after refueling outages since the SRVs are setpoint tested and postmaintenance tested on-line with normal system pressure at low reactor power or decay heat. The plant has Crosby spring-type SRVs which the licensee stated have not experienced significant setpoint drift. The staff generally agrees that the setpoint drift for the plant-specific model SRVs has not been as severe as for some other model SRVs used in the industry. The licensee further stated that the plant-specific analyses for the power uprate condition conservatively assume that only 12 of the 18 plant SRVs are operable for performing the SRV safety function. This additional margin provided in the plant-specific analyses gives reasonable assurance that the SRV setpoint drift would not result in the maximum allowable system pressure being exceeded. Therefore, the staff agrees that the licensee's proposal to not incorporate the resetting of the SRVs to ±1 percent in the plant TS is acceptable.

The NRC staff reviewed the results of the licensee's analyses, and concludes, that a setpoint tolerance increase from +1%/-3% to ±3%, and allowing up to two SRVs to be OOS was properly analyzed by the licensee and is acceptable.

TABLE 3.1

Containment Performance Results

Parameter	Current FSAR Case(1)	Updated Methods FSAR Case	Updated Methods at Uprated Power(2)(3)	Final at Uprated Power With ELLL(2)(3)	Limit
Peak Drywell Pressure (psig)	34.7	34.5	34.8	35.1	45
Peak Wetwell Pressure (psig)	27.3	30.2	30.2	(4)	45
Peak Drywell-to-Wetwell Pressure Difference(psid)	19.4	21.7	21.5	(4)	25
Peak Bulk Pool Temperature (°F)					
LOCA	220	199	204	(4)	212
Alternate Shutdown	215	—	210	—	212

(1) Analysis performed at 3462 MWt (104.2% of original rated power), dome pressure 1040 psig.

(2) Analysis performed at 3702 MWt (102 x 110% of original rated steam flow), dome pressure 1040 psig.

(3) Update (current) methods.

(4) Bounded by power uprate case.

Note: The value of P_a to be used for 10 CFR Part 50 Appendix J testing must be more than or equal to the peak calculated containment pressure resulting from any design-basis accident. To bound a potential future power uprate (110% of original rated steam flow) and possible unforeseen future plant changes, the value of P_a was conservatively chosen by the licensee to be 38 psig.

Table 3.2

LOCA Radiological Consequences

LOCATION	DOSE (rem)*	GUIDELINES (rem)
Exclusion Area:		
Whole Body Dose	5.7	25
Thyroid Dose	86.3	300
Low Population Zone:		
Whole Body Dose	4.1	25
Thyroid Dose	95.8	300
Control Room:		
Whole Body Dose	0.4	5**
Thyroid Dose	15.4	30
Beta Dose	3.5	30

* Based on 1.02 X uprated power

** This value is a limit derived from General Design Criterion 19.

FHA Radiological Consequences

LOCATION	DOSE (rem)	GUIDELINES (rem)
Exclusion Area:		
Whole Body Dose	1.1	6
Thyroid Dose	1.5	75
Low Population Zone:		
Whole Body Dose	0.4	6
Thyroid Dose	0.6	75

CRDA Radiological Consequences

LOCATION	DOSE (rem)	GUIDELINES (rem)
Exclusion Area:		
Whole Body Dose	0.03	6
Thyroid Dose	0.3	75
Low Population Zone:		
Whole Body Dose	0.02	6
Thyroid Dose	0.7	75

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Washington State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on May 2, 1995 (60 FR 21554).

In this finding, the Commission determined that issuance of this amendment would not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

(1) Letter to the NRC from J. V. Parrish, WPPSS, G02-93-180, requesting an amendment to the WNP-2 operating license and TS to increase licensed power level with extended load line limit, and a change in safety relief valve setpoint tolerance, dated July 9, 1993.

(2) Letter to the NRC from J. V. Parrish, WPPSS, G02-94-004, requesting an administrative change to the WNP-2 TS to support the power uprate, January 6, 1994.

(3) Letter to the NRC from J. V. Parrish, WPPSS, G02-94-006, "Response to Request for Additional Information, Power Uprate Review," January 6, 1994.

(4) Letter to the NRC from J. V. Parrish, WPPSS, G02-94-029, "Response to Request for Additional Information, Power Uprate Review," February 2, 1994.

(5) Letter to the NRC from J. V. Parrish, WPPSS, G02-94-103, "Response to Request for Additional Information, Power Uprate Review," May 3, 1994.

(6) Letter to the NRC from J. V. Parrish, WPPSS, G02-94-115, "Response to Request for Additional Information, Power Uprate Review," May 13, 1994.

(7) GE Nuclear Energy, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984P, Class III (Proprietary), July 1991; NEDO-31984, Class I (Non-Proprietary), March 1992; and Supplements 1 and 2.

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